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ABSTRACT

This report provides an assessment of safety issues at the Department of Energy (DOE) test and research reactors. Part A identifies six safety issues of the reactors. These issues include the safety design philosophy, the conduct of safety reviews, the performance of probabilistic risk assessments, the reliance on reactor operators, the fragmented character of the DOE management structure, and the safety implications of the current budgetary climate. Part B provides an assessment of technical issues to each of five reactors, such as: (1) Advanced Test Reactor (ATR); (2) Experimental Breeder Reactor II (EBRII); (3) Fast Flux Test Facility (FFTF); (4) High Flux Beam Reactor (HFBR); and (5) High Flux Isotope Reactor (HFIR). Each section includes the recent operating history and conclusions and recommendations on the technical issues. Over 400 documents are listed in the bibliography grouped by kind of reactor. Appendices include a "Statement of Task" by DOE and National Academy of Sciences-National Research Council and "Biographical Sketches of the Members." (YP)

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Safety Issues at the DOE Test and Research Reactors

A Report to the
U.S. Department of Energy

Committee to Assess Safety and Technical Issues at DOE Reactors
Commission on Physical Sciences, Mathematics, and Resources
and
Commission on Engineering and Technical Systems
National Research Council

NATIONAL ACADEMY PRESS
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This report has been reviewed by a group other than the authors according to procedures approved by a Report Review Committee consisting of members of the National Academy of Sciences, the National Academy of Engineering, and the Institute of Medicine.

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front cover: Removal of an irradiated target assembly from the High Flux Isotope Reactor.

back cover: The serpentine core of the Advanced Test Reactor.

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Preface

Following the nuclear reactor accident at Chernobyl in the Soviet Union, Secretary of Energy John S. Herrington asked the National Academy of Sciences and the National Academy of Engineering to put together a committee with expertise in reactor-safety-related disciplines to review safety and technical issues at the department's largest reactors—those categorized as "Class A reactors." The Academies formed the Committee to Assess Safety and Technical Issues at DOE Reactors, which began its study in August 1986.

At the committee's first meeting, DOE officials requested that the committee focus its initial efforts on the defense production reactors—the Class A reactors that produce plutonium and tritium for use in nuclear weapons. In October 1987 the committee released its first report, *Safety Issues at the Defense Production Reactors.* The report addressed specific safety and technical issues at the production reactors and provided recommendations for improving the overall structure and management of DOE's safety system.

The present volume, the committee's final report, covers the remainder of the department's Class A reactors. It addresses safety and technical issues at five federal test and research reactors—facilities operated by private contractors for DOE for purposes of scientific research, radioisotope production, materials irradiation, and the development of advanced reactor technology. The DOE test and research reactors include the Advanced Test Reactor and the Experimental

Breeder Reactor II in Idaho, the Fast Flux Test Facility in Washington, the High Flux Beam Reactor in New York, and the High Flux Isotope Reactor in Tennessee. (In 1987 a sixth test and research reactor—the Oak Ridge Research Reactor in Tennessee—was ordered permanently shut down by DOE.)

Although the original charge envisioned a study of safety issues at the DOE reactors in light of the Chernobyl accident, the committee considered it more productive, given the unique designs of the DOE reactors, to focus on issues identified during the study. Hence, the committee interpreted its charge broadly, and the scope of the committee's two reports extends beyond a narrow examination of the technical lessons of the Chernobyl accident for the DOE reactors.

In the course of its work, the committee examined extensive documentation on the test and research reactors from DOE and its contractors and conducted site visits at all five facilities. Additional meetings were held for briefings from high-ranking department officials, from officials of the Office of Management and Budget, and from contractor employees on technical subjects of particular interest. The committee very much appreciates the assistance that was so generously given.

One member of the committee, Herbert Kouts, is a long-time employee of the Brookhaven National Laboratory. Because Brookhaven is the operating contractor of the High Flux Beam Reactor, Dr. Kouts excused himself from committee discussions concerning that reactor and played no part in the formulation of those sections of the report that deal with it.

A number of events have transpired since the issuance of the committee's report on the defense production reactors that are germane to this report. First, the department has endorsed all of the committee's recommendations and has taken a number of steps to implement them. In response to the committee's observations concerning the need to strengthen internal oversight, the department's budget request for FY 1989 included large increases in funding and staffing for the Environment, Safety, and Health (ES&H) organization. ES&H is developing an overall safety objective, with assistance from an ad hoc committee, and is undertaking revision and review of the system of departmental orders. DOE has signed a memorandum of agreement with the Institute of Nuclear Power Operations (INPO) that will permit the department to draw upon the breadth of experience that INPO has garnered in evaluating commercial nuclear power reactors. The department has also informed the committee that the technical issues discussed in the first report are being aggressively pursued by DOE and its contractors. And, in response to committee recommendations concerning the need for external, independent review, the department formed an Advisory Committee on Nuclear Facility Safety. By May 1988 the advisory committee had begun meeting on a regular basis. These are important changes that will benefit all the reactors, and the committee applauds the department's efforts.

Second, as this report was going to press, Congress passed legislation that will create a permanent safety oversight board with responsibility for overseeing the department's defense nuclear facilities. Hence, the administrative structure for operating the DOE reactors will change in ways that are not yet well defined. The committee has prepared this report to focus on issues that will remain significant regardless of how the administrative structure is modified.

Third, during the course of the committee's deliberations on the production reactors, one of the test and research reactors—the High Flux Isotope Reactor at Oak Ridge, Tennessee—was shut down as a result of certain major deficiencies. The need to resolve these deficiencies has delayed restart. The situation is thus similar to that confronted by the committee in its examination of the defense production reactors; the N Reactor had been shut down for safety improvements in January 1987 following the publication of several critical reports by an outside panel of experts. (The department subsequently announced that the N Reactor would be placed in "cold standby" for an indefinite period.) The committee pointed out in its previous report that it had not reached any conclusions with regard to restart of the N Reactor. A similar disclaimer applies to this report insofar as restart of the High Flux Isotope Reactor is concerned. As noted in the committee's earlier report,

> Although the committee brings to its task a wealth of experience in the field of nuclear safety, it has neither the legal authority nor the capacity to conduct the in-depth scrutiny that would be necessary to judge the overall safety of any of DOE's reactors.

Nonetheless, it is hoped that the committee's observations will assist the department in its evaluations.

In examining the test and research reactors, the committee faced five extremely diverse facilities with different ages, missions, managements, design philosophies, and degrees of modification and upgrade. Because the department's articulation of a safety objective and of an unambiguous regulatory framework is as yet unachieved—as noted above, this effort is under way—the committee did not have the benefit of DOE benchmarks to guide in the evaluation of existing or planned operations. As a result, the committee has had to base its judgments on its experience in the field of reactor safety, calling attention to safety issues that require resolution and practices that seem to be obsolete or out of step with accepted norms. The committee hopes that the conclusions and recommendations in the report will lead to improvements at the test and research reactors in a variety of areas.

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Executive Summary

The United States operates five nuclear reactors to produce radioisotopes for medical, industrial, and military purposes; to provide neutron sources for use in scientific research; and to conduct irradiation and other experiments in support of the government's space, fusion, and advanced reactor programs. Collectively referred to in this report as the test and research reactors, these facilities are operated by private contractors for the Department of Energy (DOE). They include: the Advanced Test Reactor (ATR) and the Experimental Breeder Reactor II (EBR-II), located at the Idaho National Engineering Laboratory; the Fast Flux Test Facility (FFTF), located on the Hanford Federal Nuclear Reservation in the state of Washington; the High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory on Long Island; and the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory in Tennessee.

This report provides an assessment of safety issues at the DOE test and research reactors. It identifies technical issues that are applicable to each of the reactors and offers conclusions and recommendations relevant to their continued safe operation. It is a companion volume to an earlier committee report on the defense production reactors. As explained in that report, no attempt has been made to address whether any of the DOE reactors is "safe" or to define an acceptable level of risk for the DOE reactors. Although such matters are important, an evaluation of this kind would require a balancing of costs and benefits that extends far beyond the limited scope and capability of this committee. Rather, the report constitutes an examination of a number of generic and specific safety issues that are important to improving the safe operation and management of DOE's nuclear facilities.

The report is organized in two parts. Part A identifies six safety issues of generic relevance to two or more of the test and research reactors. These issues include the safety design philosophy of the test and research reactors; the conduct of safety reviews; the performance of probabilistic risk assessments; the reliance on reactor operators; the fragmented character of the DOE management structure; and the safety implications of the current budgetary climate for the test and research reactors. Part B provides an assessment of issues specific to each of the five reactors.

This report, like its predecessor, raises a number of safety-related issues and provides recommendations that the committee hopes will be helpful in resolving them. The principal conclusions and recommendations contained in Parts A and B of this report are summarized below. The committee encourages readers of this report to review the committee's previous volume, as many broad issues of safety management discussed in that volume are equally important at the test and research reactors.

GENERIC SAFETY ISSUES

The current system of line management of the DOE reactors comprises a fragmented collection of contractors, operations offices, and programmatic divisions in headquarters. The programmatic divisions have limited capabilities in the area of reactor safety. In the current system, the capacity to carefully balance programmatic needs against the safety needs of the reactors is modest. The structure tends to disperse responsibility for safety and seems to require decisions with safety consequences by organizations that are divorced from day-to-day safety responsibility. The previous report included a recommendation that DOE strengthen the internal organization responsible for safety oversight of the reactors—the headquarters-based Office of the Assistant Secretary for Environment, Safety and Health (ES&H). Although the committee continues to hold the view that strengthening the ES&H organization is important—and applauds the department's efforts in that direction—it has come to the conclusion that the structure for line management of the Class A reactors also needs significant improvement. DOE should examine whether line management of reactor operations ought to be the responsibility of a central reactor operations group within the department.

The FFTF was recently constructed and thus incorporates a wide variety of safety features. The other test and research reactors were designed before modern safety philosophy had matured; as a result, they lack the full complement of redundant and diverse safety systems that are provided in modern commercial power reactors. In part this may be justified by lesser stored energy, smaller inventory of radionuclides, and in general more remote siting than commercial nuclear reactors. Nonetheless, the risk profiles of the five test and research reactors do vary significantly, and DOE currently lacks a clear conceptual framework for addressing safety issues at the reactors. The department needs to formulate a safety objective for the Class A reactors and determine through careful evaluation whether the test and research reactors, as currently configured, are capable of achieving that objective. The committee does not prejudge the conclusion; it merely urges that the issue be addressed forthrightly.

Over time, the missions of some of the test and research reactors have changed, and the reactors have been modified, or reconfigured to conduct safety tests, in ways that are more in line with their revised missions than with their original ones. Some of the changes have occurred without the benefit of thorough external review. DOE needs to strengthen its requirements for the review of proposed changes to the test and research reactors. The reactor contractors who have not already done so should add independent members from outside organizations to their existing safety review committees in order to strengthen the review of safety issues.

Because most of the test and research reactors were designed many years ago and because they have been modified to operate outside their original design bases, prevention and mitigation of some potential accidents rely heavily on the proper and timely response of reactor operators. In extreme cases, operators might have to act in the presence of radiation or steam in order to prevent releases of radioactive materials to the environment. The department should examine the feasibility of installing equipment at the test and research reactors to ensure that responses to reactor accidents are not solely or primarily dependent on operator actions. At a minimum, the contractors should provide the capability to operate from remote consoles outside the reactor confinements those valves that are located within the confinements and may be needed to prevent fuel damage.

Probabilistic risk assessment (PRA) is an important methodology for understanding and ranking the relative risks of accidents at nuclear reactors. It has become an important part of modern analyses for commercial reactors. Attitudes and approaches toward PRA at the test and research reactors, however, are inconsistent, and the PRAs that have been completed to date are not persuasive. DOE should require that all of the test and research reactors conduct PRAs using state-of-the-art techniques, complemented by state-of-the-art deterministic analyses. For its part, DOE needs to acquire the capability to oversee the PRA review process to ensure that reviewers' comments are appropriately addressed. And, of course, any weaknesses in the plants that are revealed by the contractors' PRAs must also be confronted. In general, the execution of a PRA provides an important mechanism by which DOE and the contractors can gain greater insights into reactor safety systems.

This is a time of tight budgets in the federal government. The Office of Management and Budget (OMB) has adopted a strategy of requiring safety and programmatic needs to compete for the limited pool of federal dollars that has been allocated to the test and research reactors. While this strategy may be justified by the need to restrain the growth of the federal budget, it could have adverse safety implications unless special vigilance is maintained. To ensure that programmatic objectives do not intrude on the attainment of safety, the department should establish an unambiguous safety objective with clear safety requirements, and should strengthen line management to implement them. The department should also maintain effective oversight by a revitalized ES&H organization and by an external oversight committee. DOE should identify and defend requests for funding for safety needs separately from those for programmatic needs. Finally, DOE should formalize plans for the eventual retirement and/or replacement of the older test and research reactors.

TECHNICAL ISSUES AT THE FIVE REACTORS

This part of the report deals with specific issues at each of the five test and research reactors. One common theme is that, although modernization of safety analyses is occurring at some of the test and research reactors, there appears to be no central guiding philosophy. The modernization efforts are guided by the diverse attitudes and safety philosophies of the DOE contractors, rather than by any apparent central departmental policy to upgrade and rationalize the analyses. Another common theme of Part B is that some of the operating contractors have not kept up to date with safety activities and ways of thinking about safety that have evolved within the commercial nuclear industry.

Modernization of safety analyses at the Advanced Test Reactor (ATR) is proceeding along an appropriate path. However, there is a need to develop greater understanding of potential hydrogen-related challenges to the ATR confinement, and there are numerous accident analyses that remain to be completed. These analyses will be particularly intricate owing to the unique design of the ATR. Adequate resources, sufficient time, and thorough peer review must be available in order for these analyses to produce credible results. In the ATR design, a reduction in the amount of water in the experimental loops results in an increase in the criticality of the reactor (a positive void coefficient of reactivity). This means that loss of water from and/or depressurization of one or more of the loops can result in a rapid increase in power, potentially resulting in extensive core damage. The committee knows of no practical design change that would remove this vulnerability while also allowing the reactor to continue to achieve its mission. Loop operators at the ATR are typically among the least trained personnel, are isolated from the main control room, and have been involved in a number of recent incidents. The contractor and DOE need to upgrade the existing training program for experiment operators, and the program needs to include a more careful review of existing procedures based on experience operating the experimental loops. ATR management should ensure that experienced operators are on duty in the experimental loop area during each shift.

At the Experimental Breeder Reactor II (EBR-II), there are currently no plans to conduct a PRA of any kind. Although the plant has a number of passive safety features that suggest a high level of plant safety, a PRA can help to determine whether the expected strengths of the design, such as the shutdown cooling system, have unanticipated weaknesses. Even if the EBR-II design is found to perform well in transient-initiated accidents, other accident vulnerabilities may exist. The contractor should conduct a PRA that includes careful evaluation of the risk of refueling accidents, the reliability of the EBR-II reactor protection system, and the reliability of containment isolation.

The Fast Flux Test Facility (FFTF) at Hanford is in the midst of converting to the use of metal fuel. The FFTF fuel conversion presents a number of safety issues that are only partially resolved. Adequate resources and more realistic schedules may be needed to allow the completion of the necessary analytical and experimental work. Because operation of the reactor using metal fuel is a significant modification, in-depth reviews by the contractor's safety review committee, the Office of the Assistant Secretary for Environment, Safety, and Health, and the DOE Advisory Committee on Nuclear Facility Safety will be necessary. In the course of these reviews, particular attention needs to be devoted to analyses of transient over-power events and the behavior of new safety devices (so-called gas expansion modules) that will be relied upon after the conversion to metal fuels.

The contractor is also considering a project that would transform the facility into a power producer. DOE should carefully weigh whether such a venture offers sufficient benefits to justify the added cost in increased complexity and diffusion of the FFTF mission.

The FFTF contractor also needs to undertake a PRA. In light of the conversion to metal fuel, the PRA should examine the evolution of loss-of-flow and transient over-power events into core disruptive accidents using state-of-the-art methods. The use of FFTF's filtered vented containment to cope with a severe accident should be investigated using the latest information about debris coolability, steel and concrete penetration rates, radioactive source terms, and the potential for containment pressurization.

The High Flux Beam Reactor (HFBR) has been modified to operate at a power level above its original design basis. This modification has involved a significant change in the safety philosophy for the facility in that it has created the potential for fuel melting under loss-of-coolant conditions. Since many of the emergency operations include heavy reliance on reactor operators to respond to potential accidents, a realistic assessment should be made of potential doses to operators to ensure that adequate protection is provided both to individuals at the site and to the public in the event of an accident. In addition, the contractor's recent preliminary analyses of dynamic thermal-hydraulic effects during flow reversal following loss of forced flow conditions, and of potential reactivity accidents caused by the addition of light water into the in-core thimbles, should be confirmed and subjected to independent peer review. Planned improvements to provide remote reading of water level in the core at stations where light water can be added to the reactor coolant system should be promptly implemented.

The High Flux Isotope Reactor (HFIR) has been shut down since November 1986 when unexpected embrittlement of the reactor pressure vessel was discovered. A range of management deficiencies was subsequently discovered. DOE must ensure that the steps taken to correct deficiencies at HFIR continue to be effectively implemented both by the contractor and by the local DOE operations office.

The HFIR pressure vessel has become embrittled, and further operation cannot be conducted in compliance with the original criterion for ensuring pressure vessel integrity. The contractor has formulated an alternative strategy of operation, but thus far insufficient effort has been applied to a realistic assessment of the consequences of vessel failure. The contractor should reanalyze the consequences of vessel failure in light of modern knowledge of radionuclide releases and modern methods of accident analysis.

Introduction

Following the April 26, 1986, accident that devastated Unit Four of the Chernobyl Nuclear Power Plant in the Soviet Union, the Secretary of Energy requested that the National Academy of Sciences and the National Academy of Engineering review safety and technical issues at the department's Class A reactors—those capable of producing more than 20 MW of thermal power. The committee began its study by focusing on the department's defense production reactors and issued a report on that subject in October 1987. The committee then turned its efforts to a review of the department's remaining Class A reactors—the Advanced Test Reactor (ATR), the Experimental Breeder Reactor II (EBR-II), the Fast Flux Test Facility (FFTF), the High Flux Beam Reactor (HFBR), and the High Flux Isotope Reactor (HFIR). These are the test and research reactors—the subject of this report.

COMPARATIVE OVERVIEW OF THE TEST AND RESEARCH REACTORS

The test and research reactors represent three distinct groups of facilities. The first consists of EBR-II and FFTF, which are under the budgetary and programmatic jurisdiction of DOE's Assistant Secretary for Nuclear Energy. They are unique in their use of liquid sodium metal as a coolant, and are, in fact, the only liquid metal reactors (LMRs) in operation in the United States. Originally built as testbeds for the development of large-scale liquid metal reactors for use by the commercial power industry, EBR-II and FFTF are primarily used to test the effects of irradiation on developmental LMR materials, components, and fuels, and to investigate the passive safety characteristics of LMR designs. Both reactors have had shifting and highly uncertain missions since the withdrawal of federal support for construction of a demonstration LMR—the Clinch River Breeder Reactor project. Funding for the Clinch River reactor was terminated by the Congress in 1983.

The second group of test and research reactors consists of two reactors producing especially high neutron fluxes—the high flux beam and high flux isotope reactors. Both HFBR and HFIR are under the budgetary and programmatic jurisdiction of the DOE Office of Energy Research. Although HFBR and HFIR can and are used on a relatively small scale for testing the effects of radiation on materials, they were originally built for different purposes. HFBR was constructed with neutron scattering research as its principal mission, whereas HFIR was built with radioisotope production in mind. Together, HFBR and HFIR represent the bulk of the nation's investment in user facilities for neutron scattering research. As befits their status as user facilities, both reactors have separate areas set aside as experiment rooms where scientists can use beam tubes and can collect and analyze experimental data. In addition, over the last 20 years HFIR has been the primary U.S. source for a variety of high-assay radioisotopes with important scientific, medical, and industrial uses.

The ATR falls in the third group of reactors. It differs from the others in mission—it is the principal irradiation facility for the development of advanced naval reactor fuels and materials—and line responsibility for the reactor is not assigned to any single DOE program. Budgetary jurisdiction and decisions on what tests will be conducted in the reactor are the responsibility of the Deputy Assistant Secretary for Naval Reactors, while line responsibility for safety rests with the Assistant Secretary for Nuclear Energy.

RADIONUCLIDE INVENTORIES AND DECAY POWER

The Department of Energy's test and research reactors are small in comparison to defense production reactors and modern commercial nuclear power reactors. Table 1 provides a comparison of the

TABLE 1 Power Levels and Radionuclide Inventories

Reactor	Operating Power (MWt)	Fuel Mass (kg)	Radionuclide Inventory[a] (Ci)	Decay Heating Rate (MW) at Various Times After Shutdown	
				50s	10,000s
HFIR	85	11.9[b]	$4.3\text{x}10^7$	2.8	0.63
HFBR	60	12.4[b]	$2.1\text{x}10^7$	2.7	0.52
EBR-II	62.5	345[c]	$7.6\text{x}10^7$	2.3	0.59
ATR	250	39-46[d]	$2.4\text{x}10^8$	12.6	2.4
FFTF	400	2,928[e]	$3.1\text{x}10^8$	20	--
Commercial PWR	3,414	101,100[f]	$1.6\text{x}10^9$	100	26
Savannah River Reactor	2,915[g]	113,000[h]	$2.2\text{x}10^9$	154[i]	63[i]

[a] Radiologically important isotopes of Kr, Xe, I, and Cs calculated at shutdown for refueling.

[b] U_3O_8.

[c] U--An approximate value is given neglecting breeder material and test fuel.

[d] Range of U metal loadings.

[e] $(U,Pu)O_2$ including blanket material.

[f] UO_2.

[g] Power typically varies from 650 MW to 2,915 MW; all values in this row assume full power operation.

[h] Derived on the basis of 1,240 kg of U-235 in the Mark 15 charge with an enrichment of 1.1 percent.

[i] Derived from the ratio of decay power to operating power for the Mark 31 charge.

SOURCES: Alpert et al., 1986; Church et al., 1983; Petry et al., 1986; Steimke, 1986; EG&G Idaho, Inc., 1988s; Argonne National Laboratory, EBR-II Fission Product Inventory, 8100 MWD @ 62.5 MW Activity After Shutdown; Westinghouse Hanford Company, 1987x; Brookhaven National Laboratory, 1988l; Martin Marietta Energy Systems, Inc., 1987aa.

operating power levels of a typical commercial pressurized-water reactor, the defense production reactors operated by DOE at Savannah River, and the five test and research reactors. As can be seen, the test and research reactors operate at thermal power in a range from 60 to 400 MW, whereas a modern commercial power reactor and the Savannah River reactors typically have a thermal power of approximately 3,000 MW. This means that, other things being equal, the test and research reactors pose less risk than a Savannah River reactor or a commercial power reactor; the systems are at lower pressure, and there is less energy in normal operation to "drive" an accident or to disperse radionuclides.

Table 1 also shows the amounts of fuel used in the test and research reactors. In new cores, the fuel mass of the test and research reactors is more than an order of magnitude less than that of a typical commercial power reactor or a Savannah River reactor. Again, the smaller fuel mass, in conjunction with the lower total power, is indicative, other things being equal, of lesser risk.

These differences in scale are accompanied by differences in complexity. With the possible exception of the liquid metal reactors (FFTF and EBR-II), the test and research reactors are generally far less complex in design than either the defense production reactors or commercial nuclear power plants; they have less cable and piping and fewer pumps, valves, and safety systems. Simplicity is generally a desirable characteristic—assuming a streamlined system provides adequate safety protections—because of reduced aggregate probability of component failure and ease of system maintenance, control, and analysis.

If core cooling were to be lost in a reactor, radionuclides in the reactor fuel would provide the driving force, in the form of decay heat, for fuel melting and accident propagation. The rates of decay heat generation in the test and research reactor fuels are also shown in Table 1. In an absolute sense, the decay heating rates produced in test and research reactor fuel are all smaller than those produced in commercial nuclear reactor fuel. Thus, the systems needed to remove decay heat in an accident are smaller in these reactors than in commercial pressurized-water reactors or the Savannah River reactors. However, the decay heating rates *per unit mass of fuel* are larger for the test and research reactors than for typical commercial reactors. This indicates that under emergency conditions the time available to reestablish core cooling and prevent fuel damage is shorter. Hence, emergency core cooling is at least as important for the test and research reactors as it is for commercial reactors.

Finally, Table 1 shows the inventories of the more volatile radionuclides—isotopes of krypton, xenon, cesium, and iodine. The inventories for the test and research reactors are all more than a factor of 3 (and at HFBR a factor of nearly 50) less than those in a typical commercial pressurized-water reactor or one of the Savannah River reactors. Nonetheless, the radionuclide inventories in the test and research reactor fuels are not insignificant. Even the smallest of the test and research reactors contains several million curies of volatile radionuclides that must be controlled during an accident. Thus, although the potential threat at the test and research reactors may be appreciably less than at commercial reactors or at the production reactors, design and operation must be approached with care and vigilance.

Part A
Generic Safety Issues

Overview

The Department of Energy (DOE) faces a formidable challenge in ensuring the safe operation of its Class A reactors for several reasons. First, the designs of the reactors are dissimilar to one another and to commercial nuclear reactors. Hence, the methods of analysis used to evaluate the safety of either commercial reactors or other DOE reactors may need extensive modification before they can be appropriately applied to these facilities. Moreover, the extensive experience of the nuclear utility industry with commercial reactor operation has not been thoroughly applied to the Class A reactors. The department and its contractors have maintained safety largely through their own efforts without effectively drawing upon the extensive support network that exists in the commercial reactor world.

Second, the majority of the reactors were designed in the 1950s and 1960s, and problems of physical aging, such as vessel and beam tube embrittlement, are now being encountered. Moreover, in the interval since the reactors were originally built, the demands of society for safety of nuclear plants have increased. Thus, although severe fuel damage during design-basis accidents may have been an acceptable risk at the time the reactors were built, the risk associated with such an accident may not be acceptable today, particularly if protection of the public would require that operators take emergency actions in hazardous radiation and steam environments, as it might at some of the test and research reactors.

Third, the reactors are operated under a fragmented management system. Responsibility for the reactors originally rested with the Atomic Energy Commission (AEC). Many of the regulatory and associated functions of the AEC were transferred in 1974 to the Nuclear Regulatory Commission (NRC), but the responsibility for the Class A reactors was eventually lodged with DOE. Within DOE, the day-to-day management responsibility currently rests with local operations offices, while programmatic responsibility—including the responsibility to seek funding for safety programs and modifications—is assigned to several different assistant secretaries. The transfer of responsibility from AEC to DOE, and the division of responsibility within DOE, while perhaps aiding in the accomplishment of the variety of missions of the department, has impeded the development of a coherent strategy for defining, achieving, and maintaining the safety of the Class A reactors.

Fourth, as the programs have evolved, some reactors have been modified to enhance their capacity to serve their missions, and in some cases the missions have changed. For example, one of the reactors (HFBR) serving the programs of the Office of Energy Research now operates at a higher power level than specified in the original design. (The higher power level has yielded a greater neutron flux, which has increased the value of the reactor as a research tool.) One finds a similarly changing situation at FFTF. With the demise of the Clinch River Breeder Reactor program, the mission of FFTF has been redefined, and FFTF is now being used to conduct safety tests that were not part of the original mission of the reactor. Indeed, planning is under way to redirect the mission of FFTF again. (EBR-II provides yet another example of a reactor with a changing mission.)

Operating nuclear reactors in ways that depart from their original design can have serious safety implications. Although DOE has recognized this fact and has examined the safety implications of most of the changes that have been made, the fact remains that the design changes have, in some instances, reduced safety margins. (As discussed below, some important modifications to the HFIR reactor were not reviewed and approved by DOE prior to installation.)

Fifth, as the committee noted in its report on the defense production reactors [National Research Council, 1987], there has been a change in the way safety is conceived and implemented. The AEC placed great reliance on its reactor contractors and could do so because the contractors' employees were instrumental in designing the reactors and had an intimate knowledge of them. As time has passed, some of the original contractors have relinquished operation of the plants, and most of the employees who were instrumental in the design of the reactors have retired. Concomitantly, the depth and strength of the contractor staffs have diminished, establishing a need for more formal processes to ensure safety. In addition, public attitudes toward reactor safety have evolved, and this has necessitated a departure from the more personal and informal management of reactors that existed in the past. DOE has had to adapt to this change, but has done so slowly and incompletely.

Finally, as discussed in the previous report, the department has failed to articulate a coherent safety objective that clearly delineates the requirements that these reactors must satisfy. In the absence of such a clear safety benchmark, there is no well-defined basis to determine those areas in which improvement is needed. As a result, some critical decisions, including judgments as to when a particular reactor has reached the end of its effective life, are not guided by a coherent and accepted safety framework that balances the risks and benefits of particular missions.

In the report on the production reactors, the committee urged the department to address the problems that these developments have created. It was recommended that DOE establish a clearly articulated and documented safety objective, that it specify and implement orders that reflect that objective, and that it maintain vigilance to ensure the objective is being satisfied. It was further recommended that the department strengthen its internal oversight arm (ES&H) and establish strong external oversight in a new advisory committee. DOE was also urged to expedite training aimed at restoring critical safety functions and controlling critical safety parameters in the event of abnormal conditions. Finally, the department was urged to address a series of other technical issues. The department has acknowledged the validity of these recommendations and has initiated actions to respond to them. While this effort is to be applauded, the problems are deep-rooted and will not yield immediately to solutions; they will require continued effort at all levels in the department for years to come.

The committee's observations on the need for change have been reinforced by its examination of the test and research reactors. The previous recommendations are generally valid for these reactors as well. However, there are several points of a generic nature that deserve particular attention.

Conclusions and Recommendations

SAFETY DESIGN PHILOSOPHY

Conclusion: Several of the DOE's test and research reactors were designed and constructed before modern safety philosophy had matured. Some of the reactors are not protected by sufficient defense-in-depth measures to prevent certain accidents that could cause partial or extensive fuel damage.

A particular safety philosophy—termed defense-in-depth—was employed in the design of nearly all reactors that are operating today. Defense-in-depth aims to provide multiple lines of defense against damage and includes the incorporation of redundant and diverse engineered safety features that would be automatically actuated in order to ensure that accidents initiated by events with even very low probability (such as a double-ended break of a major coolant pipe) do not lead to damage. Modern commercial reactors not only have systems to control releases of radioactive materials in the event of fuel damage, they also have systems that are expected to preclude significant fuel damage.

Although FFTF and, to a lesser extent, EBR-II were designed with a high degree of defense-in-depth, accidents at the other test and research reactors involving single failures of reactor systems, such as breaks in pipes, could lead to significant fuel melting, with the potential for the release of a substantial fraction of the inventory of radioactive material in the fuel to the confinement and possibly into the environment. Because of the comparatively small inventory of radionuclides in the cores of these reactors, and the mitigative effect of the safety systems that are available, the consequences of these accidents are calculated to be small for the general public—that is, less than the dose limits prescribed by the NRC for design-basis accidents (10 CFR 100).

The HFIR, for example, does not have an emergency core cooling system capable of dealing with large pipe breaks. As a result, loss-of-coolant accidents involving breaks larger than 3 in. are estimated to lead to extensive fuel damage. In the event of such accidents, doses at the site boundary are calculated to be less than the NRC limits found in 10 CFR 100, assuming that 100 percent of the fuel is damaged during the accident but that other systems function as designed to attenuate the release of volatile radionuclides.

The approach to safety used in the design of the older test and research reactors was characteristic of the time in which they were built. By virtue of their small sizes and remote locations, it was felt that the reactors did not represent a serious hazard to the public, even if an accident were to occur involving significant fuel damage. It is not clear, however, that such an approach is acceptable today. The department needs to take appropriate steps to estimate the probability of events involving fuel damage at the test and research reactors, and ensure that those probabilities are acceptably small.

Recommendation: Once DOE establishes its new safety objectives, it should carefully evaluate the degree to which the test and research reactors satisfy them. Thereafter, DOE should reevaluate compliance with safety objectives at regular intervals.

SAFETY REVIEW PROCESS

Unreviewed Safety Questions

Conclusion: There have been instances at the test and research reactors where design changes were implemented or tests conducted without the benefit of a formal, externally reviewed safety analysis.

In its review of plant documentation at the test and research reactors, the committee found instances where design changes had been implemented or tests conducted that were outside the original design bases. For example, as discussed in greater detail below, at HFIR the contractor implemented changes to components in the reactor, without the requisite review and approval by DOE, that increased the estimated probability of a core damage accident by 2 orders of magnitude. (This problem was subsequently rectified.) Good practice dictates that formal analyses of the effect of changes to a reactor be conducted before the changes are implemented.

Regulations governing the implementation of design changes are set out in 10 CFR 50.59 for commercial nuclear power plants licensed by the NRC. These regulations, and the supporting guidance on their implementation, prescribe a process for determining whether any proposed change in operations or in the configuration of plant systems involves an "unreviewed safety question." If an unreviewed safety question is determined to exist, the plant is expected to complete and submit to the NRC for approval prior to implementation an in-depth safety analysis of the proposed change. (An NRC licensee must also follow very formal procedures, subject to audit by the NRC, for the review of changes that are found not to constitute an unreviewed safety question.)

In theory, DOE reactor contractors are subject to a similar requirement. The department's orders include a provision that requires that proposed changes be examined to determine whether any unreviewed safety question exists. If the change involves such a question, the contractor must submit a safety analysis to DOE for approval. This provision is important because it is one of the few mechanisms for assuring external review of changes to the DOE reactors. The committee found, however, that DOE has failed to provide sufficiently detailed guidance to the contractors to ensure that consistent procedures are used in analyzing proposed changes. DOE has also failed to closely audit contractor performance to ensure that unreviewed safety questions are being fully addressed.

Recommendation: DOE should strengthen its existing requirements for the analysis of proposed changes to plant systems, proposed changes in procedures, proposed experiments, or other preplanned deviations from the previously analyzed safety design basis in order to ensure that changes to reactor operations that can affect safety undergo a thorough, formal, independent safety review prior to implementation.

Safety Review Committees

Conclusion: Safety review committees have been set up by the contractors at each of the test and research reactors. These committees review plant safety, reactor operations, and the safety of experiments. In the case of the liquid metal cooled reactors, they also review experiments on the plants themselves. Because almost all the members of these review committees are in-house contractor employees, the committees may not bring sufficient independent judgment to the consideration of safety matters.

Contractors at the test and research reactors have established safety review committees whose members are drawn from the plant, the rest of the organization at the site, or an affiliated laboratory. (At one reactor (HFIR) the in-house committee is complemented by an independent review committee that draws its members from outside the contractor organization; however, this second committee was established only at the beginning of the current year and meets infrequently.) While the chairmen of some of the safety review committees are empowered to call upon expert consultants from outside organizations, in practice such advice is seldom solicited.

It is good practice for safety review committees to have a portion of the membership drawn from outside the operating company. Outside membership can help the safety review committee achieve a more balanced and acceptable judgment on technical issues, particularly on contentious subjects on which there may be differences of opinion within the operating organization itself. Independent members can also supply knowledge and experience from outside the organization and thus provide a mechanism for the cross-fertilization of ideas. Moreover, the presence of experienced independent members on a safety review committee can provide an important measure of public confidence in the integrity of the committee process.

In recognition of the important role of the safety review committees and to provide a sounder basis for audits, a set of general criteria and guidelines applicable to safety review committees would be useful. It is particularly important that the committees have the authority to set their own agenda and are free to examine safety issues of their own choosing, not solely those that are brought to them by DOE or the contractor. To ensure that the criteria and guidelines for safety review committees are implemented in a consistent fashion and the committees are working effectively, a plan for regularly auditing the committees should be developed and implemented.

Recommendations: The Safety Review Committees for the test and research reactors should be strengthened by the addition of independent members. DOE should establish criteria and guidelines for the effective operation of the committees, and should develop and implement plans for ES&H audits.

Review of Safety Tests

> *Conclusion: Although Argonne (EBR-II) and Westinghouse (FFTF) developed special procedures for their transient testing programs and had the programs reviewed internally, almost no external (outside the company) reviews were performed, other than by DOE staff.*

Safety tests are typically performed with reduced safety margins in comparison with those in place during normal plant operations. It is perhaps not surprising, therefore, that there is a history of accidents occurring during the performance of safety tests at nuclear reactors. The accidents at Windscale (England), NRX (Canada), EBR-I (Idaho), and Chernobyl (Soviet Union) all occurred during such tests. Based in part on this historical record, the committee believes that safety tests at any reactor should be subjected to particularly careful review.

The transient testing program at EBR-II is intended to demonstrate the "inherent" or passive safety characteristics of the metal-fueled, pool-type, liquid-metal-cooled reactor design of the EBR-II. By design, the series of tests involved significant decreases in plant safety margins and increased accident risks. In preparation for the test program, which began in 1984, test procedures were developed by the staff and were reviewed internally by the EBR-II safety and experiment review committees and by the ANL Reactor Safety Review Committee comprising Argonne staff assigned to other parts of the laboratory. Temporary waivers from restrictions in reactor technical specifications were submitted for approval to the DOE Chicago operations office, where they were reviewed by a safety program officer. They were also reviewed for concurrence by the Idaho operations office and the program office (Nuclear Energy) at DOE headquarters. Similar types of reviews were undertaken for the transient testing program at FFTF.

An important characteristic of the reviews at EBR-II and FFTF is that, for the most part, they were all performed internally. Virtually the only reviews of the safety tests at EBR-II and FFTF conducted by individuals outside the contractor organizations were those by DOE, which has very limited capability to perform this function.

Prior to the breakup of the AEC, safety tests of the type conducted at EBR-II and FFTF would have been required to be submitted for review by the Advisory Committee on Reactor Safeguards (ACRS). DOE could have used the services of the ACRS under the current law, but elected not to refer these tests to that body. The ACRS is made up of knowledgeable individuals from a variety of organizations and with a diversity of expertise, thus providing careful external review of technical issues. With the formation of DOE's new Advisory Committee on Nuclear Facility Safety, an option now exists for similar review of significant safety tests at the DOE reactors.

Recommendation: Proposals to conduct significant safety tests, like the transient test series conducted at EBR-II and FFTF, should be carefully reviewed in advance by the DOE Office of the Assistant Secretary for Environment, Safety, and Health and by the DOE Advisory Committee on Nuclear Facility Safety. More generally, DOE needs to establish criteria to determine the types of activities that will require review and approval by the DOE advisory committee.

PROBABILISTIC RISK ASSESSMENT

> *Conclusion: Efforts to produce probabilistic risk assessments (PRAs) for the test and research reactors are uneven, indicating that the value of a PRA in risk management is not uniformly appreciated by DOE and its contractors.*

Attitudes toward PRA at DOE and among the contractors of the five test and research reactors vary significantly. The ATR contractor is conducting a Level 3 PRA; the HFIR contractor has completed a Level 1 PRA; the HFBR contractor is about to begin a limited, low-priority PRA activity; and the EBR-II and FFTF contractors are not planning to undertake any PRA work at all. (As explained in the committee's report on the production reactors, a Level 1 PRA is limited to calculations of the probabilities of fuel damage accidents. A Level 2 PRA includes estimates of the timing, types, and amounts of radioactive materials that could be released from the facility. And a Level 3 PRA includes calculations of risks to public health and the economic consequences of potential releases.)

One of the arguments presented to the committee for not conducting PRAs for the test and research reactors was that the reactors contain small inventories of radioactive materials and the associated risk from accidental release is very small in comparison to the risk from commercial nuclear power plants. Although this assessment of risk may be accurate, the committee nonetheless believes that the insights to be gained from a PRA, in understanding plant safety and guiding decisions on plant modifications, fully warrant the effort and expense involved, even for the test and research reactors. The argument that the small size of the reactors translates into relatively little risk to public health does not alter the fact that the risk must be strictly controlled; as noted in Table 1, the radionuclide inventories in these reactors are not trivial. The PRA process provides a systematic methodology for exploring various accident scenarios and can lead to a better understanding of the importance to safety of various systems, designs, procedures, maintenance activities, and human performance, and thereby can help to guide decisions affecting those matters.

In the late 1970s, the nuclear power industry debated the relative merits of PRA and conventional deterministic analyses. By the 1980s, a general consensus had developed within the reactor community that both deterministic analysis and probabilistic risk analysis should be applied in understanding and managing the risks of nuclear reactors. The committee does not accept the argument that deterministic analyses—with their reliance upon such ill-defined, qualitative terms as "hypothetical" and "highly unlikely" accidents—provide a sufficient basis for all safety decisions. Deterministic analyses simply cannot provide the insights into the relative importance of potential accidents or contributors to accidents as those derived from the use of probabilistic methods. Both methods should be used.

The HFIR experience with PRA provides a good example of the value of PRA as a systematic evaluation tool. In 1985 an ad hoc committee at HFIR recommended certain modifications to the reactor to improve irradiation capabilities [Martin Marietta Energy Systems, Inc., 1985a]. The first phase of the project involved redesign of the reactor target tower and associated reactor components in order to permit on-line measurements of instrumented target samples during reactor operations. These modifications were being completed when the HFIR PRA was begun. In the course of conducting the PRA, it was quickly discovered that a major pathway to core damage (one referred to as "flow blockage due to the target tower centering

ring coming out of its groove") had been created directly as a result of the modifications to the target tower. Indeed, the mean frequency of core damage attributed to this scenario was calculated to be very high (2.6×10^{-2} per reactor year), and immediate action was taken to remedy the situation. An O-ring was removed from the target tower design, eliminating the scenario of principal concern. Additional modifications resulted in lowering the estimated overall core damage frequency to 5.1×10^{-4} per reactor year. While this example raises questions concerning the adequacy of the original design of the target tower and the safety review that accompanied the design modification, it also illustrates the usefulness of PRAs for evaluating proposed plant modifications and for identifying changes to the plant or to the procedures that can reduce risk.

The committee's report on the defense production reactors highlighted several features of PRAs that can add to their credibility. Two bear repeating here in connection with the test and research reactors.

First, by their very nature, PRAs rely heavily on the judgment of experts. This is one of the reasons that makes an independent peer review of a PRA essential. The credibility of any review, of course, still depends, as well, on the quality of the reviewers and the scope of their effort. In order to ensure a probing review, it is generally recognized that reviewers should be selected from diverse sources (e.g., from academia, industry, and the national laboratories). There are indications that some of the test and research reactor contractors may not be following these guidelines—in one case, purportedly because of problems arising from federal laws relating to the need for competitive bidding in federal contracting [EG&G Idaho, 1988l]. Furthermore, in the one instance to date where external reviewers have been assembled (HFIR), the committee found that the PRA contractor was not fully responsive to reviewers' comments, and there was inadequate oversight of the review by the reactor contractor and by DOE [SAROS, 1988; Pickard, Lowe & Garrick, 1988].

Second, while the overall structure of a PRA is fairly well established, there are some areas in which the methodology is still evolving. A state-of-the-art PRA would obviously encompass the latest methodological developments in these areas, which include analyses of human reliability, potential external initiators of accidents (e.g., earthquakes, high winds, fires, and floods), model uncertainties, and the proper elicitation of expert opinion. The analysis of potential external accident initiators is an essential aspect of modern PRAs, and the proper elicitation of expert opinion is of particular importance to DOE's reactors because of their unique designs. Formal, state-of-the-art methods for eliciting and using expert opinions are described in the technical literature, including a set of PRAs that is being developed for the NRC (NUREG-1150).

There is a general need for an integrated approach that encompasses the development and application of state-of-the-art probabilistic and deterministic analyses at the older test and research reactors. Consideration should be given to the development of a formal, integrated safety assessment program (ISAP), along the lines of the ISAP initiated by the NRC at the Millstone I and Haddam Neck nuclear power plants in 1985 or the ISAP-II recently proposed by the ACRS.

Recommendation: All of the DOE Class A reactors should have Level 1 PRAs. The need for Level 2 and Level 3 PRAs should be given careful consideration, particularly if a facility is expected to operate for an extended period of time, or if the insights they may provide are needed in evaluating the costs and benefits of future modifications.

These PRAs should be performed using techniques that incorporate state-of-the-art treatment of human reliability, external events, and uncertainty analysis and recent approaches to the systematic elicitation of expert opinion. They should be subjected to a high-quality peer review, and DOE should acquire the capability to oversee the PRA review process to ensure that PRA contractors are fully responsive to reviewers' comments.

DOE should also consider developing a formal, integrated safety assessment program for the older test and research reactors.

RELIANCE ON OPERATORS FOR EMERGENCY ACTION

> *Conclusion: In response to potential accidents at some of the test and research reactors, reactor operators must open valves or scram the reactor manually because there are no alternative systems to prevent core damage and fission-product release. In extreme cases, some of these operations would have to be undertaken in hazardous steam and radiation environments.*

When the HFBR is operated at 60 MW, the operators may be required to open certain valves inside the confinement in environments that could be hazardous to their health, in order to prevent potential accidents from progressing to core damage and fission product release. At the ATR, emergency procedures require operator action in order to mitigate potential design-basis core-damage accidents, again in circumstances in which the operators might have to act in a hazardous environment. Procedures involving heavy reliance on operators have also been proposed at HFIR and FFTF and approved by the safety review committees, although in both of these cases project management overruled the proposals.

In general, it is poor practice to rely solely or even primarily on operator action for essential safety functions. Indeed, it is now customary in the commercial sector to include automated systems that provide adequate margins of safety against incorrect operator action or against complete failure of the operators to act. Such systems are obviously of particular importance if operators would otherwise have to perform critical functions in environments that could be hazardous to their health and that would endanger the public if not performed successfully.

A critical lesson learned from the Chernobyl accident is the importance of the training and retraining of reactor operators—both formal and on the job. This point was emphasized in the committee's report on the defense production reactors, but it needs to be reaffirmed here since it applies to the test and research reactors as well.

Recommendation: DOE should examine the feasibility of installing equipment at the test and research reactors to ensure that responses to reactor accidents are not primarily dependent on operator action. At a minimum, DOE should ensure that reactor operators have the capability to operate from remote consoles outside the confinements. Care must be taken to ensure that remote operation of these valves cannot be accomplished prior to reactor shutdown, and more importantly, to ensure that installation of a remote console reduces rather than increases accident risks.

DOE MANAGEMENT OF REACTOR OPERATIONS

Conclusion: Management responsibility for safety within the Department of Energy is fragmented. The current division of responsibility may inhibit the formulation and implementation of coherent and uniform design requirements and operational practices.

The Department of Energy asserts that the assurance of safety is a line responsibility, subject to oversight by the ES&H organization in headquarters. The report on the defense production reactors emphasized the importance of strengthening the ES&H organization to ensure that the safety oversight function is performed vigorously. Strengthening the ES&H organization was thought to be particularly important for the defense production reactors, and we hold the same view for the test and research reactors as well. However, the committee has come to the conclusion that the structure of line management within DOE may also need to be improved.

The test and research reactors are operated by five different contractors, with responsibility for day-to-day oversight delegated to "local" DOE operations offices that report to the Under Secretary. (Largely for historical reasons, the Chicago operations office has responsibility for both the HFBR on Long Island and the EBR-II in Idaho, even though there is an Idaho operations office with responsibility for the ATR.) In addition to contractors and operations offices, there are several different divisions at DOE headquarters with programmatic responsibility for operation of the test and research reactors. Two of the test and research reactors (HFIR and HFBR) are within the programmatic purview of the Office of Energy Research. Another two (EBR-II and FFTF) are programmatically important to the operations of the Assistant Secretary for Nuclear Energy. And one reactor (ATR) supports activities that are almost exclusively part of the programs of the Deputy Assistant Secretary for Naval Reactors.

The multiplicity of contractors, operations offices, and programmatic divisions at headquarters, along with the diversity of missions of the reactors, has led to considerable unevenness in operating practices. The suitability of the existing arrangement is undermined by the absence of adequate staff in the DOE line management who are sophisticated on safety and operational matters, and by the fact that ES&H is not yet sufficiently strong to establish a unified and coherent safety strategy. In effect, the system relies almost exclusively on the skills and competence of the contractors. While the contractors are necessarily the first line of defense, the ultimate responsibility for safety must reside with DOE.

The committee thus finds a fragmented line management structure that serves to diffuse responsibility and, indeed, may result in organizations that are divorced from direct safety responsibility having to make decisions of immediate safety significance. For example, funding decisions for the reactors, including funding for safety upgrades, are made within the programmatic offices at headquarters, whereas real knowledge of the reactors resides almost entirely in a contractor responsive to a local operations office. (The head of the local operations office does have an opportunity to petition the Under Secretary on funding matters—in effect, to bypass the program offices—but this authority is seldom exercised.) Not surprisingly, perhaps, the current structure has failed to produce consistently rational approaches to prioritizing technical issues and allocating resources at the various facilities.

If true line responsibility for safety is to be realized, there may be advantages in consolidating line responsibility in a single departmental entity in charge of supervising the contractors at all DOE reactors. This activity might be assigned to an existing DOE office, or it might be established in a separate operating division reporting directly to the Under Secretary. Such a change might enable more efficient use of knowledgeable staff, help promote consistency across the department, and encourage and facilitate wider application of safety lessons among the reactors.

There are difficulties in reorganization: the reactors have diverse missions, the responsibilities for funding are diffused, and there is an historical relationship between headquarters and the local operations offices. Furthermore, the advantages to be gained from centralization of reactor operations would have to be balanced against the disruptive effects of such a change. Resolution of such organizational issues goes beyond the purview and expertise of the committee; yet, because these particular institutional questions can have a direct bearing on safety, the department should consider restructuring its management of reactor operations.

Whether DOE does or does not establish a central operations group, two other measures would be helpful in lessening the fragmentizing effects of the current structure. First, in order to assist DOE contractors in arriving at consistent and balanced allocation decisions, a clear, consistent, risk-based methodology for prioritizing technical issues needs to be developed. Second, there needs to be a better means of facilitating communication among the various DOE organizations and the reactor contractors.

Recommendation: DOE should examine whether the line management of reactor operations can and should be made the responsibility of a central reactor operations group.

DOE should develop and apply a clear and consistent risk-based methodology for prioritizing and allocating resources to safety issues at the Class A reactors.

DOE should also establish one or more annual meetings for key department and contractor personnel to explore potential solutions to safety issues at the Class A reactors.

BUDGETARY IMPACTS

> *Conclusion: Tight budgets can be expected for the indefinite future for the Department's test and research reactors, creating pressures on the contractors and the Department to postpone needed safety improvements in order to maintain existing programs.*

Representatives from OMB made clear to the committee that, insofar as the administration is concerned, the programs that support the test and research reactors will not receive extraordinary funding to respond to safety problems [National Research Council, 1988]. Rather, the department will be expected to accommodate any funding to respond to safety (or other) problems from the budget allocation that otherwise would be made available for program activities. Because the activities that are being pursued at the test and research reactors do not have high political visibility, the Congress is not likely to intervene to alter significantly the OMB budget strategy. Although the OMB approach may be justified by the need to restrain the growth in the federal budget, it must be recognized that, absent special vigilance, such a strategy could have adverse safety implications.

To the extent that funds to respond to safety concerns must be taken from program budgets, there will be understandable pressures on line management (the contractor and the DOE program offices) to ensure that the achievement of safety objectives does not have an adverse impact on program objectives. The committee has no evidence that these pressures have resulted in inappropriate actions by line management but believes that specific actions are needed to ensure that safety margins are maintained. Two actions would assist in this effort.

1. *Safety Framework.* As noted earlier, in the 1987 report on the defense production reactors, it was found that the department had failed to articulate an operationally meaningful safety objective and to establish a clear set of safety requirements. The committee's

observations as to the need for such a framework in connection with the production reactors apply with full force to the test and research reactors, although it might be appropriate to recognize the differences between the production reactors and the test and research reactors in such a framework. In the absence of such a safety framework, there might be a tendency for safety to diminish over time in the face of budgetary pressures. Each year of safe operation may inappropriately be seen to justify an incremental diminution of safety margins. Expensive safety upgrades might tend to be deferred so that research or other programmatic activities will not be reduced. Limitations on the budget thus inevitably provide an incentive to "make do" with existing systems and to avoid upgrades until absolutely necessary or, worse, until after an accident has occurred. (Deferral of maintenance and safety upgrades for cost reasons occurred at the N Reactor in the late 1970s.) The establishment of an unambiguous safety objective and of clear safety requirements would provide a benchmark by which to measure the adequacy of safety systems and operational performance. The establishment of a strong safety framework can thus serve to avoid any tendency to reduce existing safety margins at the DOE reactors.

2. *Management.* The report on the production reactors emphasized the importance of strengthening the management structure by which the department seeks to ensure safe operation. It noted the importance of the continued strengthening of the independent oversight of the reactors by bolstering the organization led by the Assistant Secretary of Environment, Safety, and Health (ES&H) and by establishing an external oversight committee. These recommended organizational changes also apply to the other Class A reactors. The ES&H organization reports directly to the Under Secretary and does not have any direct responsibility for the missions of the reactors. Vigilant examination of the safety of the reactors by this organization—coupled with participation in the department's budget process—can thus serve to ensure that programmatic objectives do not intrude inappropriately on the attainment of safety. Similarly, the examination of safety issues by an aggressive oversight group is important to ensure that problems at the test and research reactors are detected and corrected expeditiously.

Budgetary constraints are of particular concern for the older test and research reactors such as HFIR, HFBR, and EBR-II. Each of these reactors is experiencing deterioration because of aging. At both HFIR and HFBR, irradiation is embrittling the primary coolant system pressure boundaries of the reactors. Although an exemplary study of plant aging at EBR-II did not reveal any problems of immediate concern, over time more and more of the resources available for the older reactors will have to be consumed in the effort to counteract aging, simply to maintain the existing safety margins. Eventually, given the current budgetary outlook and increasing signs of facility deterioration, retirement and/or replacement of the facilities is likely to be more cost effective than continuing to struggle to meet the department's safety standards. The committee believes that it is not premature to undertake serious planning for the retirement and, as appropriate, replacement of the department's older reactors.

Recommendation: To ensure continued safe operation of the Class A reactors in the face of budget constraints, DOE should establish an unambiguous safety objective and clear safety requirements and should maintain vigorous and effective oversight of safety both by a revitalized ES&H organization and by an external oversight committee.

To assist in mitigating the competition between safety and programmatic goals, DOE's budget requests should identify and defend major safety items separately from requests for programmatic purposes.

In view of existing budgetary constraints and increasing signs of facility deterioration due to aging, DOE should formalize plans for the eventual retirement and/or replacement of the older test and research reactors.

Part B
Technical Issues at the Five Reactors

Introduction

The sections that follow provide brief descriptions of the five reactors and their recent operating histories, along with discussions of specific technical issues. The discussion of these issues in connection with any one reactor may not apply fully to the others for two reasons: diversity and inconsistency.

The degree of diversity in the designs of the test and research reactors is readily apparent from the reactor descriptions. In addition, the fact that the test and research reactors are operated by five different contractors with a range of technical and management capabilities has resulted in further diversity among the facilities.

Although some inconsistency in the application of standards may be the inevitable result of the diversity among the reactors, there is a residuum of inconsistency that reflects the department's approach to management of the reactors. DOE has not provided a common safety framework for the interpretation of its orders, resulting in the disparate implementation of DOE requirements and modern safety standards.

The chapters that follow also establish that modernization of the reactors is occurring at different paces and with different objectives; the tools that are being applied in conducting accident analyses differ markedly; and the depth of analyses accompanying modifications and other proposed changes in operations varies significantly.

Advanced Test Reactor

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Advanced Test Reactor (ATR)

The Advanced Test Reactor (ATR) is a light-water reactor with a thermal power of 250 MW. It is located at the Test Reactor Area of the Idaho National Engineering Laboratory, approximately 40 mi. west of Idaho Falls. The ATR went critical in 1967 and was raised to full power in 1969. Its basic mission is the irradiation of reactor fuels and materials, almost exclusively for the Naval Reactors Program. The reactor is also used to produce a small quantity of cobalt 60 and other isotopes for commercial use. The operating contractor of the reactor is EG&G Idaho, Inc. The ATR has a staff of 272, and a current annual operating budget of $38.5 million.

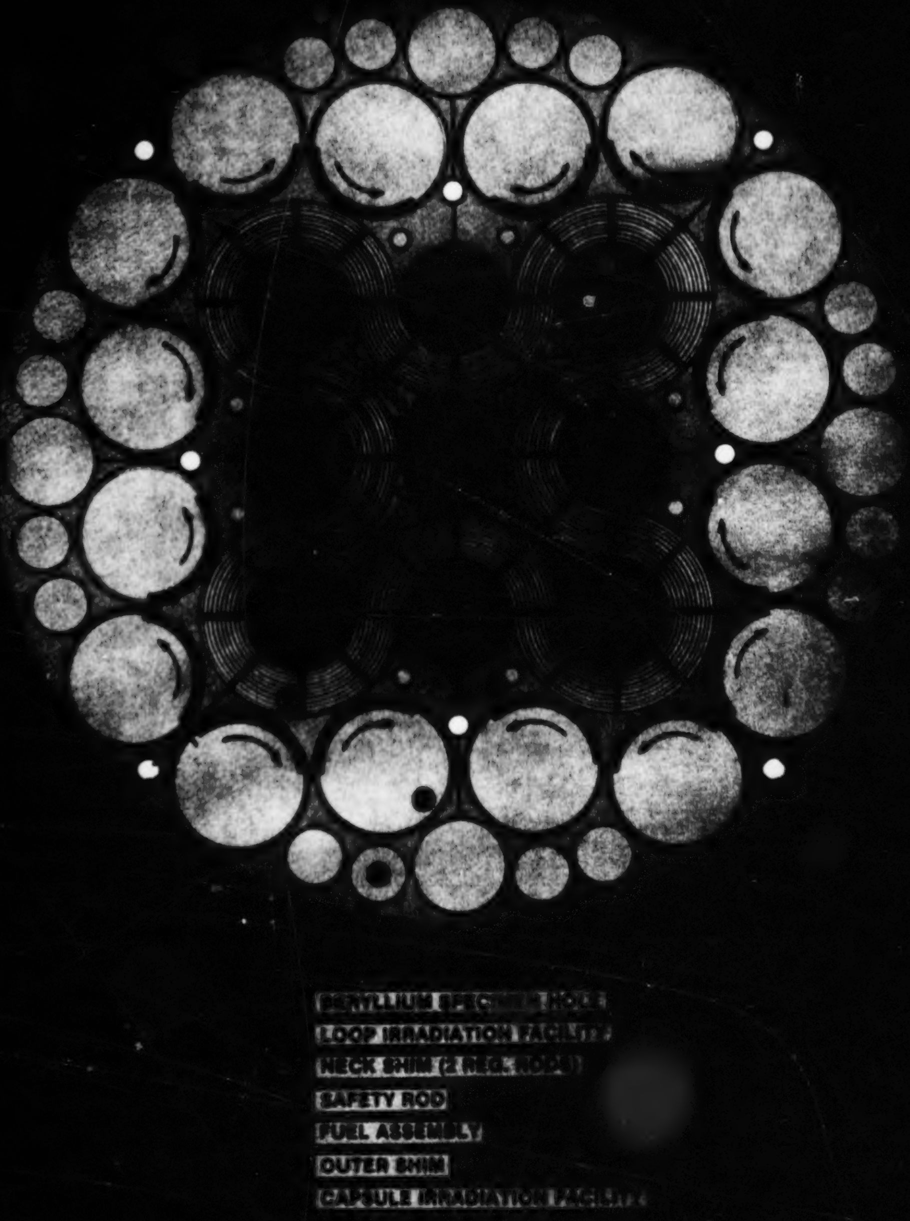
The ATR is unusual in that the fuel is not in a compact core. Seen from above, the core looks like a curvy ribbon, winding in and around a three-by-three array of irradiation positions (see ATR core cross-section print and back cover). The fuel configuration thus resembles the outline of a four-leaf clover. The total length of the ribbon of fuel around the four lobes in the core is about 11 ft. The largest dimension, diagonally across two opposite lobes, is 3 ft.

The power level of each of the four lobes can be varied from 17 to 60 MW. There are four large beryllium cylinders arranged around the outside of each lobe, and each cylinder has a hafnium absorber covering 120° of its surface. If these cylinders are rotated so that the hafnium faces the fuel, the local reactivity (and hence the local power level) is lowered. In addition, there are six smaller hafnium control rods on the inside of each lobe. These rods are used mostly to compensate for fuel burnup. In order to minimize power distortion, any individual rod is usually either fully inserted into or fully removed from the core. Finally, for shutdown purposes there are six fast-acting safety rods that also use hafnium absorbers.

There are 40 identical fuel assemblies, each with two aluminum sideplates angled at 45° to each other. The sideplates hold the curved aluminum-clad fuel plates. There are 19 plates in each assembly. The plates are 4 ft high by 0.05 in. thick, with the length of the plate arc increasing from 2.4 in. at the inner radius to 4.4 in. at the outer radius. Typical coolant spacing between the plates is 0.078 in. The innermost and outermost plates are approximately double the thickness of the other plates. The fuel is an intermetallic compound of aluminum and uranium (UAl_x). The uranium is 93 percent enriched, and there are 39 to 46 kg of U-235 in a fresh core. The maximum flux is 1.0×10^{15} neutrons/cm^2/s. Both the total flux and the energy spectrum are strongly dependent on position in and around the core.

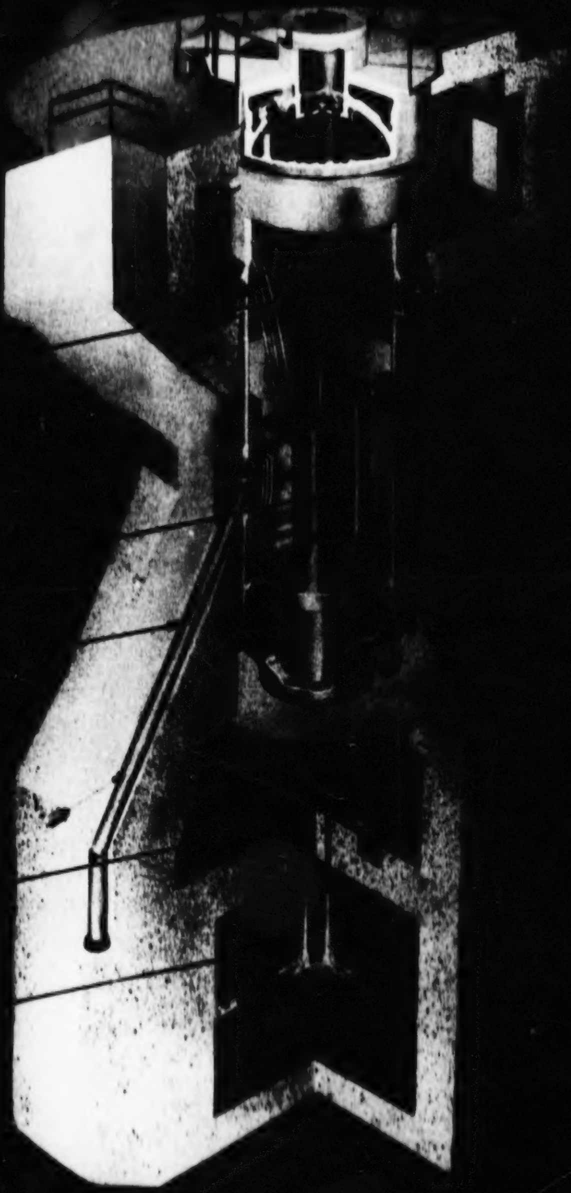
The reactor vessel is solid stainless steel, 12 ft in diameter and 35 ft high, with walls 2 in. thick. Coolant water enters the vessel through two pipes at the bottom, flows upward in the vessel outside the cylindrical tanks that support and contain the core, and enters the open part of the vessel above the core. The coolant flows downward through the core at 47,000 gpm. The temperature rises from 125°F at the vessel inlet to 167°F at the outlet, as the pressure drops from 355 psi to 255 psi. The fuel is designed to retain its integrity at surface temperatures up to 368°F, above which plate buckling initiates. Four outlet pipes at the bottom of the core take the coolant up to vessel outlet ports, and then through five parallel heat exchangers. Two 10,000-gal. demineralized water storage tanks on the ATR site are used to store primary system makeup water. Additional raw water storage exceeding 1,000,000 gal. is also available for emergency addition. The secondary system cycles water at 31,000 gpm from the heat exchangers to a six-bay cooling tower. To minimize accumulation of solids in the secondary due to evaporation, water is continually purged and replaced. The make-up water, which constitutes almost 10 percent of the flow, is supplied from the 600,000-gal. inventory in the cooling-tower cold well.

Each of the nine irradiation positions in the core contains an independent pressurized water loop. These loops are used as test facilities, and each is the responsibility of an operator who is not in the


BERYLLIUM SPECIMEN HOLE
LOOP IRRADIATION FACILITY
NECK SHIM (2 REG. RODS)
SAFETY ROD
FUEL ASSEMBLY
OUTER SHIM
CAPSULE IRRADIATION FACILITY

ATR CORE CROSS SECTION

ATR Vessel
and
Shielding



reactor control room. The test facilities, the major reason for ATR's existence, are used primarily for testing naval reactor fuels and materials. Each loop has its own shielded basement cubicle containing its own pumps, heat exchangers, pressurizer, and demineralizers. Altogether the loop support facilities occupy two basement floors around the reactor and below the control room.

Other available test facilities include capsule irradiation thimbles. There are 36 small thimbles, less than 1 in. in diameter, located in and around the core; these are used mainly for long-term irradiations of materials samples. Twenty other holes in the core, up to 5 in. in diameter, are located outside the beryllium cylinders. Thirty-four additional spaces are available in two capsule irradiation tanks that hang on the outside of the core-reflector tank. There are also facilities for gamma irradiation in the fuel storage grid of the ATR storage canal.

The ATR has a confinement system that is designed to withstand a pressure equivalent to 7.5 in. of water. It encloses the reactor and its control room, as well as the operations areas for the test facilities and the associated elevators and stairwells. The only way to vent confinement pressure is through the stack. Because the exhaust system has no filtering capability, the stack is designed to close if high radiation is detected in it. Although the primary system can be depressurized from outside the confinement to allow the low-pressure emergency coolant to be added to the system, there is limited capability to monitor post-accident conditions inside the confinement after evacuation.

The original ATR design assumed a 20-year lifetime. Monitors are being used to gather data on the aging of all major reactor systems. Present practice is to replace the core internals of the reactor about every 8 years. There is a program underway to produce a plan aimed at extending the reactor's lifetime to 45 years.

RECENT OPERATING HISTORY

The ATR was in its 80th cycle of operation in April of 1988. (One cycle consists of the installation, burnup, and removal of a fuel loading.) The operating schedule and annual loadings of the reactor are determined by the classified test program developed by the Office of Naval Reactors. Operating cycles are typically either of 15-day or 35-day duration, and between runs the plant is shut down for a period of 4 to 7 days. In February 1989 the contractor plans to shut the reactor down for 40 days to upgrade and replace the existing instrument panels in the control room.

Since reactor start-up the most significant safety-related operating events at ATR have included unexpected cracking of the beryllium reflector, which led to a 3-month shutdown in 1972, and two overpressure incidents (in 1972 and 1977) in which design pressure limits were slightly exceeded. In addition, in 1986 a workman dropped foreign material into the reactor vessel that could not be recovered. However, the event was not deemed to be significant enough to preclude continued safe operation.

ATR has been continuously upgraded since 1969. Over 50 major modification projects have been undertaken at a total cost of $80 million (current dollars). These projects have included upgrades to the plant protection system, emergency cooling water injection system, security system, simulator, control room, process instrumentation, fire protection system, seismic instrumentation, and experimental loops. The contractor has plans to undertake an additional 18 major modifications at a cost of $22 million during the FY 1989-1992 period.

TECHNICAL ISSUES

Safety Analyses

> *Conclusion: Modernization of safety analyses of the ATR reactor is well under way and generally on the right track. Efforts are aimed at improving understanding of severe accident behavior, particularly in relation to limited core damage scenarios, and developing a risk-based management system to support future operation and management of the facility. These severe accident analyses are intricate, and so a full understanding of the relevant physical phenomena will not come easily. The potential threat to the confinement from accidents involving hydrogen generation is not yet adequately understood.*

Several activities are under way using modern techniques to improve the understanding of the ATR, including reevaluation of selected reactivity accidents and loss-of-coolant accidents (LOCAs), examination of severe accident behavior and potential threats to confinement, and execution of a PRA. These activities, as well as normal operational safety and engineering reviews for the ATR, benefit significantly from the strong technical capabilities offered locally by EG&G Idaho's staff.

Reactivity and Loss-of-Coolant Accidents

Accident evaluations performed as part of the original design basis for ATR made the bounding assumption that certain accidents (e.g., large LOCAs) resulted in 100 percent core melt [Phillips Petroleum Co., 1965a,b; Idaho Nuclear Corp., 1967]. Assuming a confinement leak rate of 10 percent per day, projected doses at the site boundary were shown to be less than 10 CFR 100 limits. Accident analyses are currently being performed to evaluate more realistically the extent of core damage associated with potential accidents at ATR. For example, early results of the analysis of a reactivity transient resulting from a hypothetical 2-in. break in one of the experiment loops show a peak power of 1,000 MW and some core damage [EG&G Idaho, Inc., 1986a]. In the case of LOCAs, unpublished analyses by the contractor indicate that operation with a distribution of 30, 25, 25, and 20 MW in the four lobes of the core would be expected to produce fuel damage only in the high power lobe. However, a LOCA at power levels closer to ATR's 250-MW rated power, or with different power distributions, might damage the entire core. These examples indicate that further analyses are required to better define the envelope of limited core damage accidents at ATR.

Because the reactor has a high power density and the potential for experiencing flow instability [Aerojet Nuclear Co., 1975], its response to potential reactivity accidents and LOCAs is sensitive to the specific ways in which power and coolant flow vary during transients. In the ATR, power and flow are coupled because of reactivity feedbacks that derive from expansion of the fuel and the reactivity effects induced by the ATR experiment loops. Furthermore, because of the fuel's serpentine geometry and the asymmetric distribution of power among the four lobes of the core, spatial effects would be expected to play a significant role in determining the course of accidents. These factors mean that modern analytical tools need to be applied with special care and in full knowledge of the limits of their applicability. Provided that sufficient care is taken in applying these tools, they can assist in developing a more definitive understanding of potential conditions that could lead to core damage in the ATR.

Severe Accident Behavior

Severe accident analyses at ATR involve the examination of conduction cooling mechanisms and coolant flow through the narrow gaps that exist within the ATR fuel assemblies. Unpublished analyses described by the contractor to the committee in January 1988 do not appear to adequately consider the potential expansion and distortion of overheated assemblies during severe accidents. Moreover, the analyses do not include sufficient consideration of the possibility that molten debris could relocate in the core; steam flow could move molten material upward inside the voided assembly channels. (This phenomenon is termed "flooding.") These criticisms of the ATR accident analyses serve to illustrate the difficulty of the effort and underscore the need for adequate resources, time, and peer review to achieve reliable analyses of severe accident behavior. They also highlight the need for explicit discussion in the contractor's PRA of how uncertainties have been treated.

Potential Threats to the Confinement

Potential accidents in the ATR that might result in molten fuel interacting with coolant would also benefit from thorough analysis and review. Data on fuel-coolant interaction are available from tests conducted in another reactor (SPERT), but they should be reviewed with care before being applied to ensure that the operative parameters of the tests truly correspond to ATR characteristics. The analysis of potential molten fuel-coolant interactions in ATR should also include an up-to-date examination of vessel loading and fluid-structural material interaction problems. The contractor is now attempting to determine whether these latter considerations may be amenable to bounding analyses.

The examination of fuel-coolant interactions and potential piping failures is also important in considering the potential for hydrogen generation, detonation, and combustion. While the ATR confinement is relatively voluminous, current analyses of potential accidents involving hydrogen generation are "nonmechanistic" [EG&G Idaho, Inc., 1988k]. Thus, analyses to date have assumed varying amounts of core melt, a single melt temperature, and a single particle size to derive a "bounding" hydrogen source term. Only diffusion has been considered in analyzing hydrogen transport within the confinement. Condensation of steam has not been included. A severe hydrogen burn could conceivably produce confinement leak rates that significantly exceed the 10 percent per day rate of confinement leakage that the contractor has shown will restrict releases to values below 10 CFR 100 limits. In view of this, the committee believes that the examination of potential accidents involving hydrogen generation should be considered a high priority. The contractor is planning to examine in greater detail such issues as potential hydrogen release rates, release locations, the extent of natural circulation and forced convective flow, and steam condensation. This examination should be comprehensive in scope in order to ensure that the analyses of the potential for hydrogen generation at ATR are credible.

Probabilistic Risk Assessment

Following the Chernobyl accident, the DOE Idaho operations office formed a safety assessment review group to assess the available safety information on the ATR and to evaluate the need for any additional analyses. This group recommended that a PRA be considered for the ATR [DOE, 1987]. EG&G recognized that a PRA could be used to enhance the operation of ATR in many different ways, and decided to undertake a complete Level 3 PRA. (EG&G has some experience in supporting probabilistic risk assessments for commercial reactors.) The contractor expects to use the PRA to develop a risk-based management system that will facilitate the following:

- definition of the dominant accident sequences for the facility, including "external" events;
- evaluations of the sensitivity of existing safety-related support systems;
- evaluations of deficiencies in and potential interactions among safety-related systems;
- evaluations of potential upgrades and modifications to improve the safety performance of the facility and establish a system for prioritizing proposed changes;
- evaluations to improve human performance in operations, maintenance, training, and the development of emergency procedures;
- establishment of a basis for plant life extension; and
- evaluations of new experiments or new modes of operation that could have a significant impact on the safety performance of the facility.

As indicated earlier in the report, independent peer review can help to ensure that accident analyses and PRAs are of the necessary high quality. To date there has been little external input to the ongoing severe accident analysis efforts.

Recommendation: Severe accident analyses of potential hydrogen-related challenges to the ATR confinement should be given a high priority, and the contractor should explicitly address how analytical uncertainties have been treated in the ATR Level 3 PRA. DOE needs to provide adequate time and resources for the ATR accident analyses and PRA in order to ensure that they result in credible, high-quality assessments of plant risk.

Experimental Loop Operations

> *Conclusion: A reduction in the amount of water in the reactor's experimental loops results in an increase in the criticality of the reactor. Therefore, loss of water from and/or depressurization of one or more of the experimental loops can result in a reactivity transient, potentially resulting in extensive core damage. This situation calls for particularly vigorous attention to the safe operation of the experimental loops. Well-written procedures and well-qualified and supervised experimental loop operators are required to prevent reactivity accidents.*

The close neutronic coupling of the experimental loops with the ATR reactor means that improper operation of experiments can have an impact on reactor safety. This accentuates the importance of ensuring adequate qualification and supervision of the personnel who operate the loops. During the committee's visit to the ATR, however, the experiment operators on a particular shift had very little experience. Moreover, subsequently obtained reports of unusual occurrences at ATR revealed errors in the operation of the experimental loops—at least one of these incidents, had it occurred during power operation, could have significantly affected the reactivity of the reactor. These facts highlight the need to ensure that experienced experiment operators are on duty during every shift, as well as the need for closer management audits, root-cause analyses, and periodic retraining of experiment operators. The latter is especially important after unusual occurrences in which loop operators have been involved. The need for careful supervision of loop operations may become even more immediate a problem as more experienced operators leave (through retirement or for other reasons) and greater reliance must be placed on newly qualified personnel.

Recommendation: The existing training program for experiment operators should be strengthened on the basis of a careful review of operating experience on the experimental loops. The training program should include emphasis on corrective actions in response to unusual occurrences in order to ensure that procedures for loop operations are clear and up-to-date. Management should also attempt to ensure that there are experienced operators in the experimental loop area on each shift.

Experimental Breeder Reactor II

Experimental Breeder Reactor II (EBR-II)

The Experimental Breeder Reactor II (EBR-II) is a sodium-cooled fast reactor with thermal power of 62.5 MW. It is operated by the Argonne National Laboratory (ANL) and is located on the ANL-West site at the Idaho National Engineering Laboratory (INEL). EBR-II is the only test and research reactor that also generates electricity, supplying 14 to 15 MW electrical output for the INEL grid. Steam from EBR-II is also used to heat the ANL-West facilities. At start-up in 1964 the mission of the reactor was to prove the feasibility of a metal-fueled liquid metal-cooled reactor (LMR) in a breeder cycle. Within a few years the equivalent of five cores of fuel was cycled from the reactor through an adjoining reprocessing facility and back to the reactor.

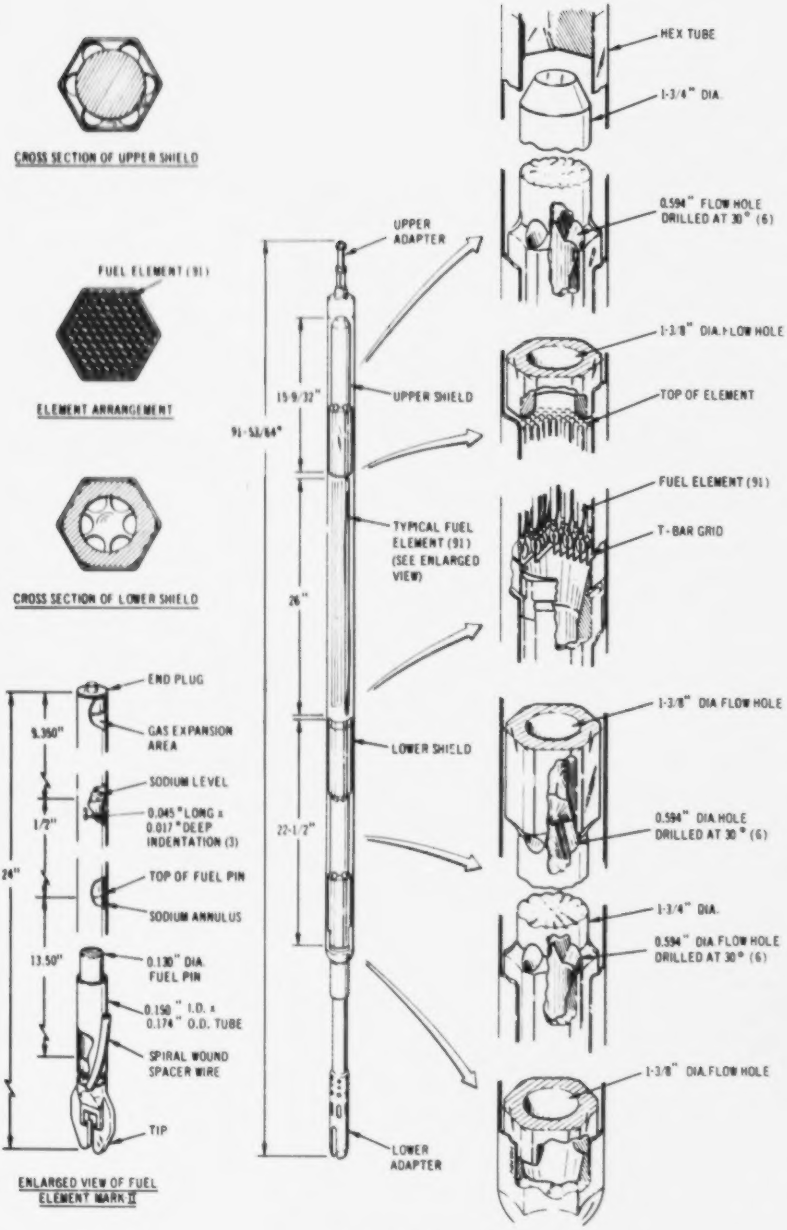
In 1968-1969, the reactor was converted to a fast-reactor irradiation test facility. Fuels and materials that are to be irradiated in EBR-II are placed in core subassemblies essentially identical to the subassemblies holding the driver fuel elements. An EBR-II subassembly is a long hexagonal tube about 2.3 in. across. The reactor core region consists of 127 subassemblies standing in a hexagonal array. Driver fuel is in approximately 57 of them, and control and safety rods use up another 11. The remaining 59 are free to hold specimens for irradiation. The subassemblies for irradiation tests are carefully located in the core; many can hold up to 91 specimens.

The EBR-II fast neutron flux ranges from 2.3 × 10^{15} neutrons/cm^2/s at the core center to 0.9 × 10^{15} neutrons/cm^2/s at the core outer periphery. (Fast neutrons are defined as those with kinetic energy above 0.111 MeV.) Mean neutron energy at the core center is about 0.4 MeV.

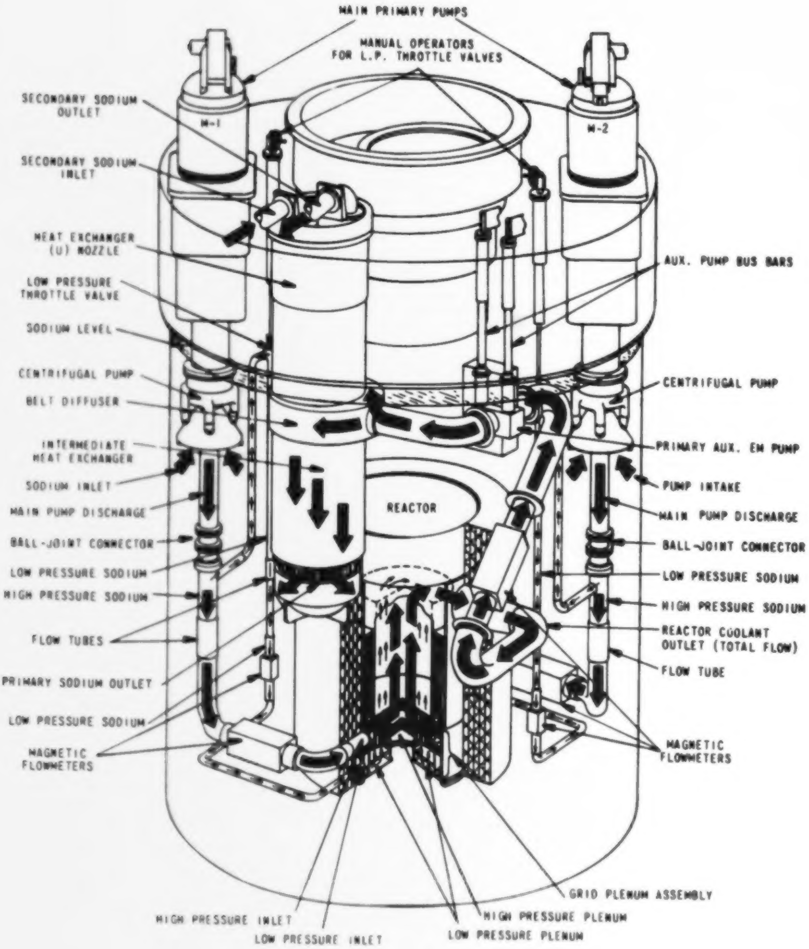
More recently, the EBR-II has been utilized to support the base technology program for developing small LMRs. Hence, the two main purposes of EBR-II are (1) demonstration of the inherent safety and shutdown heat removal capability of the LMR concept, and (2) irradiation of fuels and materials. As of May 1988, the staffing level was 253. The annual operating budget has been approximately $25 million in FY 1985, 1986, and 1987.

EBR-II is the only pool reactor of the five research reactors considered in this report. In an LMR pool reactor, the core, reflector, blanket, neutron shield, primary pumps, primary piping, intermediate heat exchanger, and in-vessel fuel handling equipment are all submerged under the molten sodium of the primary system. The primary system is cooled at the intermediate heat exchanger, which transfers heat to sodium in the secondary system. The secondary system eventually transfers heat to water and saturated steam in modular evaporators and superheaters, producing superheated steam that drives the turbine-generator and produces electricity for the INEL grid.

The EBR-II reactor has a number of attractive passive safety characteristics. Because of the pool and guard vessel surrounding the reactor vessel, the likelihood of LOCAs at EBR-II is extremely remote. A decrease in sodium density in EBR-II (such as from boiling) results in a more stable reactivity response than in larger sodium-cooled reactors. Analyses by the contractor demonstrate that if a gas or vapor bubble, with dimensions the width of an assembly or greater, were introduced into the sodium at the most adverse location in the core, it would produce a negative reactivity effect, tending to shut the reactor down. Because of the favorable reactivity behavior of the design, and the large margin between operating temperatures and fuel failure that is associated with the use of EBR-II metal fuels (Mark II U-Fs fuel with type 31 stainless steel cladding and Mark III U-Zr fuel with Type D-9 stainless steel cladding), the reactor is able to respond to loss-of-flow and loss-of-heat-removal transients without fuel damage, even without scramming the reactor. (Indeed, this facet of the design has been demonstrated in tests.) A natural-convection


CROSS SECTION OF UPPER SHIELD
FUEL ELEMENT (91)
ELEMENT ARRANGEMENT
CROSS SECTION OF LOWER SHIELD
END PLUG
9.350"
GAS EXPANSION AREA
SODIUM LEVEL
1/2"
0.045" LONG x 0.017" DEEP INDENTATION (3)
24"
TOP OF FUEL PIN
SODIUM ANNULUS
13.50"
0.130" DIA. FUEL PIN
0.150" I.D. x 0.174" O.D. TUBE
SPIRAL WOUND SPACER WIRE
TIP
ENLARGED VIEW OF FUEL ELEMENT MARK-II
UPPER ADAPTER
91-53/64"
15-9/32"
UPPER SHIELD
TYPICAL FUEL ELEMENT (91) (SEE ENLARGED VIEW)
26"
LOWER SHIELD
22-1/2"
LOWER ADAPTER
HEX TUBE
1-3/4" DIA.
0.594" FLOW HOLE DRILLED AT 30° (6)
1-3/8" DIA. FLOW HOLE
TOP OF ELEMENT
FUEL ELEMENT (91)
T-BAR GRID
1-3/8" DIA FLOW HOLE
0.594" DIA HOLE DRILLED AT 30° (6)
1-3/4" DIA.
0.594" DIA FLOW HOLE DRILLED AT 30° (6)
1-3/8" DIA FLOW HOLE

Mark II Driver Fuel Subassembly


MAIN PRIMARY PUMPS
MANUAL OPERATORS FOR L.P. THROTTLE VALVES
SECONDARY SODIUM OUTLET
M-1
M-2
SECONDARY SODIUM INLET
HEAT EXCHANGER (U) NOZZLE
LOW PRESSURE THROTTLE VALVE
SODIUM LEVEL
CENTRIFUGAL PUMP
BELT DIFFUSER
INTERMEDIATE HEAT EXCHANGER
SODIUM INLET
MAIN PUMP DISCHARGE
BALL-JOINT CONNECTOR
LOW PRESSURE SODIUM
HIGH PRESSURE SODIUM
FLOW TUBES
PRIMARY SODIUM OUTLET
LOW PRESSURE SODIUM
MAGNETIC FLOWMETERS
AUX. PUMP BUS BARS
CENTRIFUGAL PUMP
PRIMARY AUX. EM PUMP
PUMP INTAKE
MAIN PUMP DISCHARGE
BALL-JOINT CONNECTOR
LOW PRESSURE SODIUM
HIGH PRESSURE SODIUM
REACTOR COOLANT OUTLET (TOTAL FLOW)
FLOW TUBE
MAGNETIC FLOWMETERS
REACTOR
GRID PLENUM ASSEMBLY
HIGH PRESSURE INLET
HIGH PRESSURE PLENUM
LOW PRESSURE INLET
LOW PRESSURE PLENUM

PRIMARY COOLING SYSTEM

driven, heat-removal system is capable of rejecting decay heat from the plant without electric power.

The likelihood of an accident progressing to severe damage of the Mark II or Mark III fuel in EBR-II appears to be quite small because of the plant's passive safety characteristics. Collapse of the entire core during an accident and its subsequent reassembly in a uniform mass of molten fuel, as is assumed for the purpose of analyzing a hypothetical core disruptive accident in EBR-II, is an even more remote possibility. Moreover, energy absorbing structures have been provided around the vessel to contain the energy release in an accident of this type. EBR-II also has a containment building that provides added assurance that a large release of radioactivity to the environment can be prevented, provided that the building is sealed (i.e., "isolated") upon the initiation of an accident (see below).

EBR-II is now used to develop and demonstrate advanced metallic fuels, a key element of ANL's integral fast reactor program. (In fact, unlike FFTF, it has never used an oxide fuel as the driver fuel, although oxide fuels have been irradiated in it in the past.) In the reactor's early years, reloads of driver fuel came from reprocessed EBR-II spent fuel elements. After the processing facility was shut down in 1969, EBR-II used a fuel fabricated to mimic the product of the processing facility. The fuel was 95 percent uranium and 5 percent "fissium." The uranium in this admixture was enriched to 67 percent U-235. The fissium was roughly equal parts molybdenum and ruthenium, with small concentrations of rhodium, palladium, zirconium, and niobium. (The advantage of using such alloys is that they exhibit less swelling than unalloyed uranium.) The fuel pins contain sodium in the gap between the fuel slug and the cladding to transfer heat efficiently to the cladding. The cladding gap accommodates swelling of the fuel during irradiation.

More recently, EBR-II has been loaded with so-called binary metal fuels composed of uranium and zirconium. These fuels are more economical to fabricate than fissium fuels, and they can operate at higher temperatures and to greater burnup. Tests are now being conducted on more advanced, ternary metal fuels that are composed of uranium, plutonium, and zirconium.

The EBR-II staff recently undertook a review of plant aging to identify mechanisms that could potentially interfere with the continued operation of the plant. A number of potential problems were identified, including the possibility of sodium penetrating and swelling graphite-filled canisters located in the reactor head and thermal shield. None of the problems that were identified appeared insurmountable. The EBR-II plant life extension study is an excellent example of how to avoid future problems by systematic examination and proper planning. Similar studies at the other Class A reactors might provide similar benefits.

In January 1988, DOE announced that it was rescinding plans to shut the reactor down in 1993. The present plan calls for EBR-II to continue operation into the 1990s, completing the current fuel development program, demonstrating on-site fuel recycling, conducting further tests of passive safety using metal fuels, and irradiating materials for the space reactors program.

RECENT OPERATING HISTORY

EBR-II nominally operates on a 10-week cycle. Ten full-power weeks are followed by a 1-week turnaround time for refueling and minor plant maintenance. The downtime is also used for discharging and replacing irradiation samples. In addition, the plant is shut down for 4 to 6 weeks each year for more comprehensive modification, maintenance, and inspection.

The operating schedule calls for the plant to be available for operation approximately 80 percent of the time. The actual achieved capacity factor has climbed steadily in the last 5 years, from 67 percent in 1983 to 81.3 percent in 1987.

EBR-II registered 43 unusual occurrence reports (UORs) in the last 5 years (1983-87). There is no single major problem causing the UORs, although the general area of primary sodium flow control is a frequent contributor. Four of the UORs were caused by problems with primary flow indicators in the plenum. Several others involved reactor trips from momentary or spurious primary flow indications. Another general cause of UORs seems to be the jamming of moving parts owing to sodium/sodium oxide buildup. This has been responsible for two cases of shaft binding on a primary pump. (During the 1988 annual maintenance shutdown, the pump experiencing this problem was refurbished.)

There have been 15 major modifications to the EBR-II plant since startup in 1964. These have included modifications to such major systems as the EBR-II fuel elements, fuel and reflector subassemblies, control rods, and shutdown coolers for the primary coolant system. Four additional major modifications are planned. One is a change of the fuel composition and fuel-element cladding; another is a change in subassembly design to accommodate fuel burnup to 20 percent; the third involves installation of a developmental delayed neutron detection system for a joint DOE/Japanese program of oxide fuel performance testing; and the fourth is the implementation of an automatic plant start-up capability using digital-computer control equipment.

TECHNICAL ISSUES

Organizational Structure

> *Conclusion: Quality assurance responsibilities at EBR-II are mixed with the operations function, rather than being separated in an independent organization.*

The benefits of providing a separate management chain for quality assurance are generally accepted elsewhere in the DOE system. Such a structure can facilitate safety by providing oversight of line operations personnel. EBR-II does not have a management structure with an independent quality assurance group with a safety mission separate from experimental and operational goals.

Recommendation: The contractor should examine whether to reorganize to strengthen the safety and quality assurance functions at EBR-II.

Probabilistic Risk Assessment

> *Conclusion: There are currently no plans to perform even a Level 1 PRA for the EBR-II plant.*

In Part A, the committee recommended that, at the least, a Level 1 PRA should be undertaken for each Class A reactor. Although the EBR-II plant has a number of attractive passive safety features that serve to reduce the risk from loss-of-coolant and certain transient-initiated accidents, a PRA could nonetheless provide important insights. For example, the operating contractor needs an understanding of the features of the plant that have the greatest safety significance in order to ensure that resources applied to modifying the plant are properly allocated. In performing a PRA for EBR-II, the following should be carefully evaluated:

- The risk of refueling accidents. Because much of the refueling activity at EBR-II occurs under sodium, the operators cannot observe what is happening. A number of incidents and problems have occurred during the refueling process, including dropped subassemblies and a hydrogen explosion in the interbuilding coffin (a structure for transporting irradiated assemblies out of the containment).
- The reliability of the reactor control systems. There is a history of control rods sticking at EBR-II. A second control system, the safety-rod system, is primarily for use during fuel handling, when the control rods are removed. The reliability of the control systems that provide reactor protection should be examined both for normal operation and during refueling operations.
- The reliability of containment isolation. The reactor operates in a purge mode in which air is drawn into the containment by the ventilation system, provides cooling to the instrument thimbles and to the reactor shield, and after filtering is released to the discharge stack. In the event of an accident, valves would have to be closed to achieve isolation.

Recommendation: The committee recommends that a Level 1 PRA be performed for the EBR-II plant and that refueling accidents and the reliability of the reactor protection system and containment isolation be carefully examined in the context of the PRA.

Acoustic Monitoring

Conclusion: The EBR-II reactor has acoustic monitoring equipment installed to monitor vibrations in pumps and components, but the system is not currently in use.

The analysis of acoustic signals is used at nuclear power plants to identify incipient safety problems, such as vibrations in pumps, crack initiation, leakage, loose parts, vibrating core internals, and flow blockage. Loose-part monitors, for example, are required for commercial nuclear power plants. Although acoustic monitoring equipment was installed in EBR-II in conjunction with an earlier program to develop a vibration-monitoring capability, the staff familiar with the equipment are no longer employed at EBR-II, and the equipment is not being used.

Recommendation: The contractor should survey the use of acoustic monitoring systems at other reactors to determine the best use of the system currently installed in EBR-II or whether installation of an alternative system would have safety benefits.

Fast Flux
Test Facility

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Fast Flux Test Facility (FFTF)

The Fast Flux Test Facility (FFTF) is a 400-MWt liquid metal reactor (LMR). It is located at the Hanford Nuclear Reservation near Richland, Washington. Since its start-up in 1980, FFTF has been operated by Westinghouse Hanford Company (WHC). WHC also operated the Hanford Engineering Development Laboratory until June 1987, when DOE consolidated its contracts for operation of the various nuclear facilities at Hanford—including FFTF, the N Reactor, and the Purex Chemical Processing Plant—under WHC.

The primary mission of the FFTF has been the irradiation of materials and reactor fuels. A number of natural circulation cooling tests have also been performed, and the contractor plans to use the reactor to further investigate the passive safety features of LMRs. In FY 1988 the operating budget of the reactor was $41 million and the staff numbered about 375.

To date, FFTF has relied almost exclusively on oxide fuels, although a metal fuel assembly was inserted in the reactor beginning in 1986 as an initial experiment to evaluate the FFTF metal fuel concept. Conversion to a full metal core is scheduled to be completed by 1992. The oxide fuels currently in use are mixtures of PuO_2 and UO_2, with both natural and depleted uranium. The percentage of PuO_2 ranges from 22.5 percent to 29.3 percent in FFTF fuel assemblies. The plutonium is 88 percent Pu-239, with less than 1 percent Pu-241 and the remainder Pu-240. The total weight of fissile material in a 74-element FFTF core is about 560 kg, approximately 21 percent of the total oxide fuel weight.

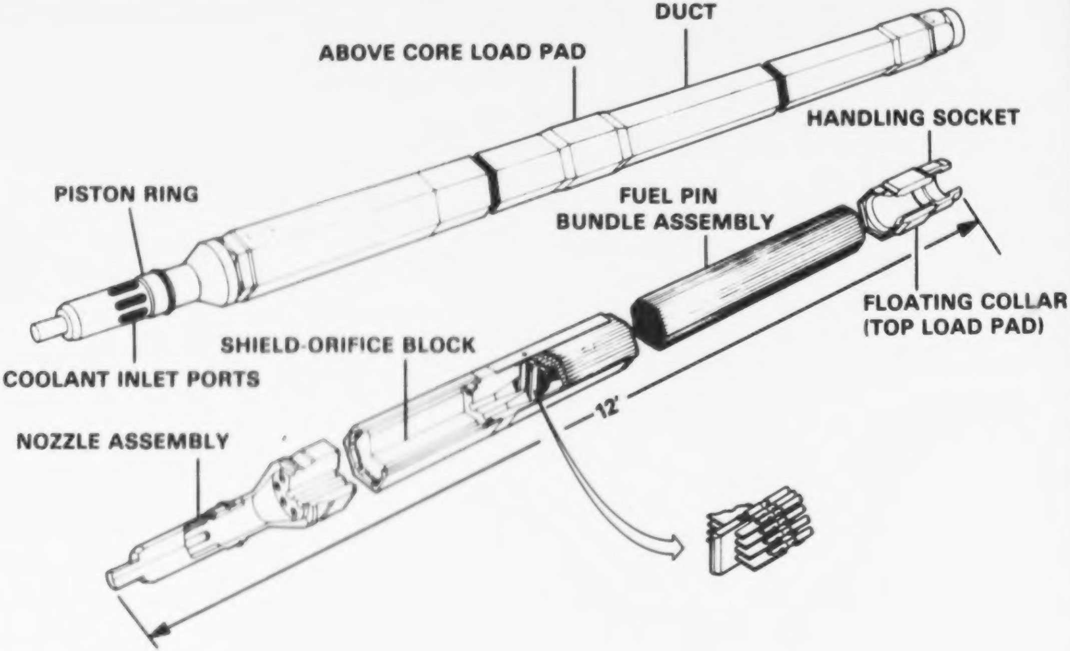
The total neutron flux at FFTF ranges from 7 × 10^{15} neutrons/cm^2/s at the center of the core to 4 × 10^{15} neutrons/cm^2/s at the periphery. The mean kinetic energy of the neutrons is 0.2 MeV; 65 percent of the flux has kinetic energy above 0.1 MeV. The planned conversion to metal fuel will decrease the flux somewhat and increase the mean kinetic energy of the neutrons.

The reactor core consists of 91 twelve-foot long hexagonal tubes, or assemblies, standing vertically in a hexagonal array. Each assembly consists of a hexagonal flow duct surrounding an internal structure. Most of these assemblies contain fuel pins; in a typical core loading, 74 of the 91 are used to hold driver fuel pins. Of the remaining 17 assemblies, 3 are safety rods, 6 are control rods, and 8 are reserved for a variety of irradiation tests. The hexagonal assemblies are each 4.6 in. across and 12 ft long, giving the total array a diameter of 4 ft. The reactor fuel is contained in a 3-ft section of the pin, providing the FFTF with a core that is 3 ft in height by 4 ft in diameter. The 8 assembly positions available for irradiation testing can be instrumented with electrical and pneumatic leads that penetrate the reactor head.

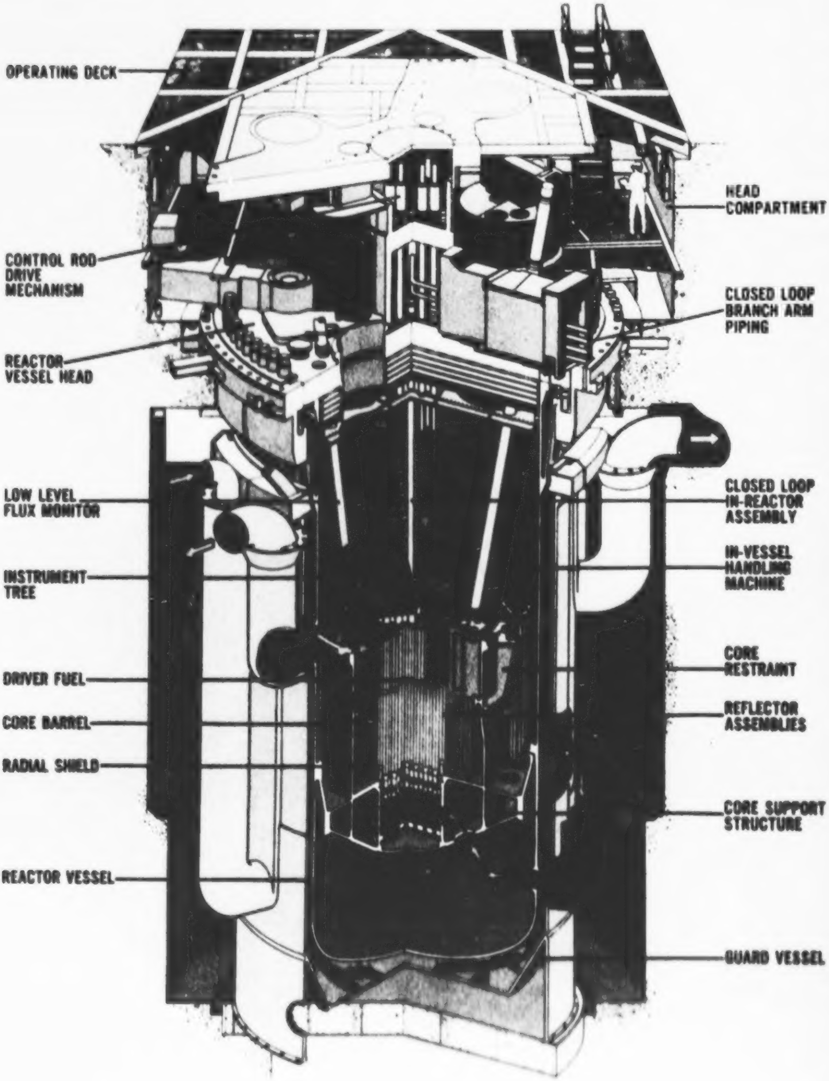
The FFTF cooling system has three primary loops that circulate liquid sodium through the reactor vessel. (Only one loop is needed for adequate core cooling; three loops provide redundancy and greater assurance of the cooling function.) Each loop has its own pump, heat exchanger, and secondary loop; only the reactor vessel is common to all three. There are 128,000 gal. of sodium in the primary system, flowing at 43,500 gpm. The temperature in the reactor vessel rises from 680°F at the core inlet to 938°F at the outlet. Each primary loop exchanges heat generated in the core with an independent secondary loop. Each of the secondary loops has its own pumps, and each has 4 air-blast (or "dump") heat exchangers. The FFTF design was reviewed by the NRC staff prior to reactor start-up.

A series of tests conducted in July 1986 demonstrated a degree of passive safety with a modified FFTF core. The test series culminated in a test in which, with the reactor operating at 50 percent power, all the coolant pumps were shut down in order to evaluate the core's response to a simulated loss of electric power to the pumps. These tests were conducted without scramming the reactor and without operator intervention. For the purposes of the tests, 9 of the Inconel

DRIVER FUEL ASSEMBLY


DUCT
ABOVE CORE LOAD PAD
HANDLING SOCKET
PISTON RING
FUEL PIN
BUNDLE ASSEMBLY
FLOATING COLLAR
(TOP LOAD PAD)
SHIELD-ORIFICE BLOCK
COOLANT INLET PORTS
12'
NOZZLE ASSEMBLY

FFTF REACTOR


OPERATING DECK
HEAD COMPARTMENT
CONTROL ROD DRIVE MECHANISM
CLOSED LOOP BRANCH ARM PIPING
REACTOR VESSEL HEAD
LOW LEVEL FLUX MONITOR
CLOSED LOOP IN-REACTOR ASSEMBLY
INSTRUMENT TREE
IN-VESSEL HANDLING MACHINE
CORE RESTRAINT
DRIVER FUEL
REFLECTOR ASSEMBLIES
CORE BARREL
RADIAL SHIELD
CORE SUPPORT STRUCTURE
REACTOR VESSEL
GUARD VESSEL

reflectors immediately outside the core were replaced with so-called gas expansion modules (GEMs). A GEM is essentially an inverted test tube with argon gas compressed at the top. With the pumps operating and providing full flow, a standing column of sodium is maintained in the bottom of the tube. The column of sodium keeps the argon compressed at the top of the tube and serves as a neutron reflector. When the pumps stop, such as during the loss-of-flow tests, the pressure drops, the argon expands, and the sodium is driven down and out of the tube. This effectively removes the reflector, allowing more neutrons to escape the core, adding negative reactivity and thereby tending to shut the reactor down. The need for such a large negative reactivity insertion to ensure safe shutdown upon loss of flow conditions results mainly from the need when using oxide fuels for a very rapid decrease in power (and coolant temperature) to prevent sodium boiling.

In conducting these tests, the contractor was limited by the fact that the temperature sensors in FFTF are located 5 ft above the fuel. Under natural circulation conditions (but not under forced flow conditions), the coolant takes a few seconds to flow from the fuel to the sensors, causing a time delay in the readout of core temperature. A safety limit of 1,074°F was set as the maximum allowable temperature at the sensors. In a test at 50 percent power and 100 percent flow, the maximum temperature measured was 950°F at 100 s, and was falling past 750°F at 400 s. Computer models indicate that without GEMs a similar core temperature would be reached in a loss-of-flow test starting from 5 to 10 percent of full power, and that the sodium in the primary system would approach boiling if the test were started from 35 percent of full power. The GEMs were inserted in the reactor only for the few weeks during which the loss-of-flow tests were conducted; that particular core configuration was never taken to full power.

There is no separate backup heat removal system at FFTF. Tests in 1980 established that the FFTF design affords sufficient natural convection heat removal capability to assure core cooling for all accidents analyzed in the safety analysis for the facility.

FFTF has plans to shift from oxide fuels to a "binary" metal fuel. The new binary metal fuel will consist of 90 percent uranium (with enrichments in the range of 26.5 to 33.8 weight percent) together with 10 percent zirconium. The cladding material will also be changed, from 316 stainless steel to HT-9 ferritic-martensitic alloy, a material that promises to solve the problem of radiation-induced swelling that has been experienced in assembly ducts and fuel cladding used in EBR-II. The core restraint built into FFTF for use with oxide fuels will inhibit radial expansion of the metal fuel, thus reducing the negative reactivity provided by expansion of the fuel assemblies, so the regular use of GEMs is under consideration.

The addition of a privately financed power generating capability at FFTF (referred to as a "power addition") is under study. The proposed power addition includes use of steam generators formerly designated for the now defunct Clinch River Breeder Reactor project. Use of two of the secondary loops at FFTF for power generation could yield about 110 MWe.

RECENT OPERATING HISTORY

FFTF began full-power operation in April 1982. As of this writing, it is in its 10th cycle of operations. Early operating cycles were devoted to characterization of the behavior of the mixed-oxide driver fuel and to test irradiations of LMR materials, components, and assemblies.

As discussed above, the reactor was subjected to a series of transient tests during cycle 8 (February to July 1986) to investigate passive safety features of the reactor's design. Beginning with cycle 9, an extensive large pin mixed oxide fuel test program was begun. In late 1986, metal fuel testing was also initiated in preparation for conversion to metal fuel.

Since 1983 the reactor has had an improving capacity factor, averaging fewer than three unplanned scrams per year. There has been only one unplanned scram or forced outage since June 1986; the reactor operated for 18 consecutive months from July 1986 to January 1988 with none at all.

The number of plant modifications that were at some stage of completion at FFTF averaged about 325 throughout 1987, but was down slightly during the first quarter of 1988. From the beginning of full-power operations, there had been a history of increasing cesium activity in the primary sodium coolant owing to cladding and fuel pin failures associated with experimental irradiations conducted at the facility. However, recent installation of a cesium trap appears to have brought the problem under control. At times, the backlog of corrective maintenance items has been extraordinarily large (1,149 items in February 1987), but this backlog was steadily reduced during 1987 until the number of outstanding requests for corrective maintenance stood at 496 at the end of March 1988.

Examination of unusual occurrence reports suggests that the greatest problems are associated with operation of the in-vessel handling machine, the bottom loading transfer cask, the zero time outage equipment (equipment for automatically transferring essential safety systems to DC batteries during loss of offsite power), and various remote handling and cooling equipment within the interim examination and maintenance cell. The latter happens to be the tallest hot cell in the nation. The most significant safety-related operating events include the following:

- In May 1982 an error was made in a computer program associated with operation of the in-vessel handling machine. The program misidentified a position in the core that was scheduled for refueling. This resulted in the inadvertent withdrawal during refueling of a control rod rather than an experimental test assembly. No criticality occurred, and the event was within the design basis of the plant.
- In June 1982, with the reactor shutdown, a primary system pump running on a pony motor seized up because of the buildup of sodium or sodium compound deposits on the pump shaft.
- In November 1984 cavitation-induced erosion of a duct on an electromagnetic pump, caused by operating the pump at excessive flow rates, led to a sodium leak.
- In October 1985 a bottle of helium used to cool specimens within a materials open test assembly was inadvertently replaced with a bottle of argon, causing overheating and damage to some of the specimens.

Although FFTF has a reactor simulator on site, the existing one is recognized as having limited capability. Westinghouse Hanford plans to request additional capital equipment funds during FY 1988-1990 to upgrade the FFTF simulator. Other planned modifications include the power addition; upgrades to the fire detection system; in-containment equipment for dealing with by-product tritium that would accompany future insertion of a fusion materials open test assembly into the core; back-up power for the zero time outage (ZTO) busses to permit repairs to the ZTO while the reactor is operating; and additional upgrades to various fuel handling equipment.

TECHNICAL ISSUES

Conversion from Oxide to Metal Fuel

Conclusion: The projected conversion of FFTF from mixed oxide fuel to metal fuel presents a number of safety issues that are only partially resolved. Resolution of these issues will require intense analytical and experimental effort, and the existing schedule for completing them is very demanding.

Because DOE intends to withdraw funding for additional fabrication of oxide fuel, uninterrupted full-power operation of FFTF beyond 1991 will require conversion from oxide fuel to a new metal fuel. The DOE objective in converting to metal fuel is to provide continued support for the nation's LMR development program.

Planning for the conversion of FFTF to metal fuel entails considerable analytical and experimental research, and an intensive program is under way. The analytical program includes design and safety evaluation of the core, fuel assemblies, and pins, while the experimental work (in progress and planned) includes pin tests, prototype assembly tests, and qualification tests in the reactor. The final design will depend heavily on the results of a core demonstration experiment that will be discharged from the reactor in early 1991. At that time there will be about 25 metal fuel assemblies in the FFTF core—a limit that derives from current test procedures and that cannot be exceeded without formal approval of an addendum to the final safety analysis report (FSAR) for the facility [WHC, 1977 as amended].

The metal fuel under consideration will not be greatly different from fuel currently being used in EBR-II, for which there is a growing body of test data and operational experience. Moreover, the FFTF metal core is expected to have steady-state mechanical and thermal-hydraulic properties that are similar to the present FFTF oxide core. But there are a number of significant differences between the FFTF metal and oxide cores. Lifetime fuel pin performance will not be the same, several reactivity effects will be significantly different, and reactor transient behavior will be modified.

The different reactivity effects will be caused mainly by the harder neutron spectrum (higher average neutron energy) with metal fuel. The harder spectrum will result in a smaller change in reactivity with fuel burnup, higher neutron leakage from the core, reduced control rod effectiveness, reduced negative Doppler feedback, and increased positive sodium void feedback. Moreover, the change from fissile plutonium to fissile uranium yields more delayed neutrons, which means that the kinetic sensitivity of the core to reactivity changes will be lower. (The reactor's response rate directly depends upon the ratio of the control change to the fraction of neutrons that are delayed. As this ratio is reduced, the reactor is slower to respond to control changes.) All these effects will have impacts on both transient and steady-state reactor operation.

The FFTF staff has identified five areas of "technical challenge" that require special attention in planning the fuel conversion [WHC, 1988a]. Some of these are more directly relevant to safety than others.

First, postulated accidents involving loss of flow without scram (LOFWOS) could yield some sodium boiling and fuel damage with the new fuel. The contractor's analysis suggests that adverse impacts can be mitigated by the use of GEMs, which are designed to introduce negative reactivity automatically during flow coastdown [see, e.g., WHC, 1988a, 1988g-i, 1988l]. However, the contractor believes it may be necessary to restrict the normal operating power level of the reactor in order to obtain an adequate margin of safety between the predicted maximum temperature of the sodium coolant and the sodium boiling temperature. An alternative under consideration is to modify the structural design of the core to permit greater thermal expansion during transients.

Second, another category of accidents could also produce some sodium boiling and fuel damage. These are known as transient overpower without scram (TOPWOS) accidents, and involve an addition of reactivity while the reactor is operating at nominal full power. The reactivity addition, which might be the result of the withdrawal of a control rod or the consequence of a seismic event, causes power to increase above 100 percent. The scram function on increasing power is then assumed to fail. Hence, power continues to increase, resulting in a mismatch between reactor power and coolant flow, leading to sodium boiling and fuel damage.

The GEM system would not be helpful in this case because coolant flow would continue during such an accident. Restriction of the operating power level of FFTF, as well as structural design changes, may be necessary to mitigate the potential effects of transient over-power events.

It is not yet clear which of these two classes of accidents (LOFWOS or TOPWOS) represents the worst case for determining design limits, though the GEM system is thought to render loss-of-flow accidents relatively harmless. Analysis is continuing, and transient tests of fuel pins in the Transient Reactor Test (TREAT) facility in Idaho will be very important in verifying fuel performance during potential FFTF transients.

Third, axial growth of metal fuel pins may limit the degree of burnup (maximum exposure in the reactor) that metal fuels can achieve in FFTF. Recent data from EBR-II suggest that at 10 percent burnup elongation of the fuel column may be somewhat larger in FFTF than was originally projected [WHC, 1988g].

Fourth, there may be an incompatibility between the new fuel and new cladding materials, which could provide a potential source of fuel failure during transients. The eutectic temperature (i.e., the temperature at which the fuel begins to attack the cladding material in which it is encased) for HT9 clad metal fuel was only recently found to be 725°C. The contractor believes that there is sufficient evidence to demonstrate that cladding penetration at this temperature would be very slow. Furthermore, while the rate of penetration increases exponentially above the eutectic temperature, the contractor believes that there are a range of reactivity management measures (e.g., use of GEMs, limits on operating cycle length, or use of fixed burnable shims) that can be instituted to preclude reaching fuel temperatures during an accident that would result in cladding penetration. However, Argonne has unconfirmed data that indicate that the eutectic temperature may actually be 700°F [ANL, 1988]. Further experiments and analysis are necessary to establish the eutectic temperature and thus to determine whether there is a need to impose new limits on reactor operation in order to account for this effect.

Finally, there is some evidence of strain in metal fuel pins removed from EBR-II that appears to be the result of trapped fission gas [WHC, 1988g]. This raises the possibility of fuel deformation or cladding failure after prolonged exposure. If confirmed, this effect would limit the expected lifetime of the new metal fuel.

Recommendation: Adequate resources and realistic schedules for analytical and experimental work must be established to resolve safety issues in support of the conversion of FFTF to metal fuel.

Approval of an addendum to the FSAR, covering operation of FFTF with metal fuel, should be contingent on in-depth reviews by the contractor's Safety Review Committee and the recently formed DOE Advisory Committee on Nuclear Facility Safety.

Gas Expansion Modules (GEMs)

Conclusion: GEMs were demonstrated to be effective safety devices in loss-of-flow tests conducted at reduced power (50 percent or less) in the FFTF oxide core. They may provide a large margin against sodium boiling for loss-of-flow accidents in a metal core, but there are some aspects of GEM behavior that need to be carefully analyzed. GEMs will not be helpful in potential transient over-power accidents in which full sodium flow continues.

The FFTF contractor has demonstrated that some measure of protection can be obtained against accidents involving LOFWOS by using the GEMs described above. Tests in 1986 showed that after shutting off the pumps with the reactor at 50 percent power, 90 percent of the negative reactivity effect of nine GEMs in the oxide core was effective within 20 s [see, e.g., WHC, 1987g]. (The test could not be conducted at higher power levels with the oxide fuel, because under certain hypothetical conditions boiling could occur.) With metal fuel, the contractor expects the GEMs to provide a significant reduction in reactivity during the most crucial phase of an unprotected loss-of-flow event, and thereby to ensure that peak sodium temperatures are maintained well below boiling.

The safety of full-power operation with GEMs in the core needs careful analysis and review before GEMs are inserted for routine operation with metal fuels. If sodium flow were suddenly restored during a loss-of-flow event, the GEMs could conceivably cause a reactivity increase and a possible power overshoot. This possibility deserves careful analysis by the contractor.

Recommendation: The contractor should perform improved computer simulations of GEM behavior in loss-of-flow and other potential transient events, and should examine the possibility of unexpected reactivity additions associated with use of GEMs before reinserting GEMs in the reactor.

Severe Accident Analyses

Conclusion: Severe accidents in FFTF have not been assessed using state-of-the-art methods developed since the reactor began operation. Uncertainties in post-accident heat removal, in the evolution of fission products from molten core debris *(radioactive source term), in energetic core disassembly, and in containment integrity need to be reexamined using up-to-date methods so that better estimates of the radiological consequences of severe accidents can be made.*

The committee has reviewed aspects of the original safety review for FFTF in order to understand the role that severe accident analysis will play in the development of an addendum to the FSAR for metal fuel. The original safety analyses for FFTF were all based on oxide fuel.

The FFTF contractor takes the view that a core disassembly accident is essentially impossible, because no reasonable scenario leading to such an accident has been found. Nevertheless, to be "conservative," a severe accident was analyzed in the FSAR by assuming such an accident and by assuming worst cases (e.g., all of the heat was assumed to be transferred to the FFTF concrete) for the events that would follow. The result was a hypothetical accident producing a maximum kinetic energy estimated at 150 MJ. It was judged that the reactor vessel and external coolant system could withstand an accident of this magnitude.

The NRC staff concurred in the contractor's assessment, but raised questions about the coolability of molten core debris [NRC, 1978, 1979; DOE, 1987; WHC, 1981]. In particular, the staff argued that hydrogen explosions or long-term pressurization during a worst-case accident might result in containment rupture. Three major recommendations were made. These concerned (1) availability of decay heat removal by natural circulation, (2) improvement of containment margins, and (3) assurance of piping integrity.

The first recommendation was satisfied by the contractor by demonstrating the establishment of natural circulation cooling in FFTF. The second recommendation resulted in the installation of a major system for filtered venting of the containment in the event of a severe accident. And the third recommendation, which concerned piping integrity, led to programs and systems for ultrasonic testing, sodium leak detection, and materials surveillance. Certain aspects of ultrasonic testing were ultimately dropped as impractical, but the remaining recommendations were carried out.

Meanwhile, more advanced methods for severe accident analysis were under development elsewhere. In particular, advanced methods of severe accident analysis for LMRs were developed in conjunction with safety analyses for the Clinch River Breeder Reactor project. Such phenomena as aerosol generation and radionuclide release rates, which are important for filter venting strategies at FFTF, were extensively examined for the Clinch River reactor. Moreover, since the completion of the FFTF safety analysis, the light-water reactor safety community has developed more sophisticated methods for understanding the penetration of steel by core debris and the intricacies of hydrogen combustion. Even though these methods have yet to be applied to FFTF, the FFTF contractor has told the committee that it has no plans to conduct a probabilistic risk assessment of FFTF.

It may be possible to establish that a core disruptive accident is a low probability event at FFTF because of the core design and because a loss-of-heat-sink accident at FFTF is sufficiently improbable. Such analyses have been performed, for example, for advanced LMRs that are currently being designed under DOE contract. Nonetheless, a Level 2 PRA would enable the contractor to compute the probabilities of events leading to loss of coolable geometry in the core and potential failures to containment that might result from core disruptive accidents.

Recommendation: The FFTF contractor should use state-of-the-art analytical methods to examine the possible evolution of loss-of-flow or transient over-power events into energetic core disruptive accidents, for cores containing partial or full loadings of metal fuel. The contractor should undertake a Level 2 PRA to assess the risks associated with these events. Severe accidents should be investigated using the latest information about debris coolability, steel and concrete penetration rates, radioactive source terms, and the potential for containment pressurization.

FFTF Power Addition

> *Conclusion: The power addition under consideration at FFTF poses a number of safety issues that are not considered in the current FSAR and its updates. Although a "safety assessment" of the power addition has been published, it has received limited technical review and does not fully resolve a number of questions.*

In support of the conceptual design and technical feasibility studies for a power addition at FFTF, the contractor has prepared a "safety assessment" document [WHC, 1987s]. The document aims to address the impact of the proposed power addition on existing safety analyses and the variety of new potential accidents that would be created by the power addition. The latter include an increased potential for sodium-water reactions, steam line breaks, loss-of-feedwater accidents, overcooling and undercooling events, and the effect of some "external" events, such as tornados and earthquakes, on the reconfigured plant.

The committee believes there are a number of aspects to the proposal that require careful analysis and review. These include the proposed mode of operation with dual control rooms, control room habitability in the event of a sodium fire, potential nonuniformity of flow in the loops, secondary-side transients, and greater potential for sodium-water reactions. If the project moves forward, a detailed design review and an addendum to the FSAR should be required. Since conversion to metal fuel is demanding extensive use of the contractor's resources, it is conceivable that undertaking another sizable project at the same time could detract from both efforts. Indeed, the power addition represents a fundamental new departure for FFTF and raises questions as to whether the new direction offers sufficient benefits to justify the cost in increased complexity and diffusion of mission.

Recommendation: The preliminary design of the proposed power addition should receive thorough review by the newly formed DOE Advisory Committee on Nuclear Facility Safety before a detailed design phase is initiated. If a final design is undertaken, it should also receive internal and external safety review. DOE should carefully consider the wisdom and timing of undertaking a power addition at FFTF.

High Flux Beam Reactor

High Flux Beam Reactor (HFBR)

The High Flux Beam Reactor (HFBR) is operated by the Brookhaven National Laboratory (BNL), which is located on Long Island, some 50 miles east of New York City. The reactor uses heavy water (D_2O) as coolant, moderator, and reflector. At start-up in 1965, the nominal operating power was 40 MW; an upgrade in 1982 raised the power level to 60 MW. The primary mission of the HFBR is basic scientific research, particularly neutron scattering experiments using the reactor's external thermal neutron beams. Other purposes include isotope production, neutron activation analysis, and materials irradiations.

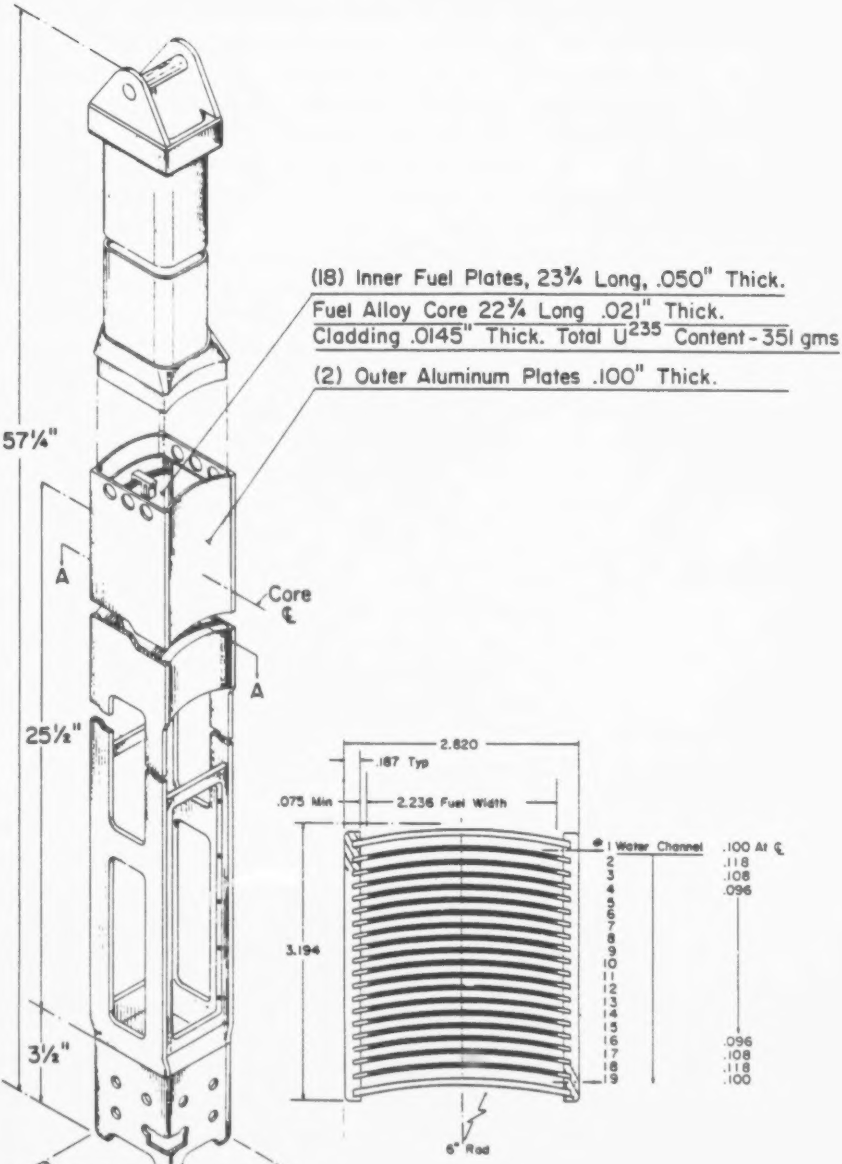
The HFBR is staffed with 65 people. It had an FY 1987 operating budget of $10.5 million, some 18 percent of which was dedicated to the operation of security programs. Construction costs of these security programs, which were installed to meet DOE requirements, were $1.9 million in FY 1986 and $1.5 million in FY 1987. An additional $1.4 million is budgeted for FY 1988.

The reactor fuel consists of U-235 contained in a cermet (*cer*amic plus *met*al) made of U_3O_8 and aluminum. The cermet is 37 percent U_3O_8 by weight, and the uranium is 93 percent enriched. A fresh core contains a total of 9.8 kg of U-235. The fuel is fabricated into plates, rather than pins, and a typical fuel plate measures 23 in. by 2.4 in. by 0.05 in. After final cold-rolling, the plates are curved to a 6-in. radius by being pressed against a curved die. The plates are clad with aluminum and mounted in a fuel element assembly, also made of aluminum. There are 18 fueled plates per element, typically spaced 0.102 in. apart. Each fuel element is approximately 3 in. square by 5 ft long. Twenty-eight such elements form the HFBR core. The active core volume is 23 in. high by 19 in. in diameter.

The reactor incorporates several unusual design features in order to provide external beams of thermal neutrons. One design goal was to have the thermal neutron flux maximized outside the core, where beam tubes could intercept the flux and channel it out to the experimental facilities. To this end, the core sits in a bath of heavy water contained within the reactor vessel. The heavy water that surrounds the core acts as both moderator and reflector; it thermalizes the fast flux, and reflects some of it back into the core. The power level is controlled by absorber blades that mask the core from the bath; inserting the blades reduces neutron reflection from the heavy-water moderator, causing the reactor to go subcritical, which in turn causes power and the average core temperature to fall. To stabilize power at a lower level and temperature, the blades must be returned to the blade position that produces criticality at the new lower temperature. In normal operation the flux of thermal neutrons is larger in the reflector region than in the core, while the fast flux peaks in the core. The peak thermal flux is 1.05×10^{15} neutrons/cm^2/s.

The heavy water used as moderator and reflector is also used as coolant. The primary cooling system contains 10,000 gal. of heavy water, flowing at 18,000 gpm. About every 18 months, one-third of the total heavy-water inventory is removed and exchanged for heavy water having a lower tritium concentration and higher purity. The purpose is to keep personnel exposures to tritium as low as reasonably achievable. Detritiated heavy water is currently obtained from Savannah River at subsidized rates. After FY 1989, however, these supplies will no longer be available. BNL is investigating the use of Canadian sources to detritiate HFBR heavy water. The cost for such services is estimated to be between $500,000 and $1 million per year.

Coolant velocity through the fuel elements is almost 40 ft/s. The temperature of the coolant rises from 130°F at the inlet to 151°F at the outlet. There are two primary loops, each with its own pump and heat exchanger. Heat is dumped to light-water cooling towers. The pressure in the primary system is much greater than pressure in

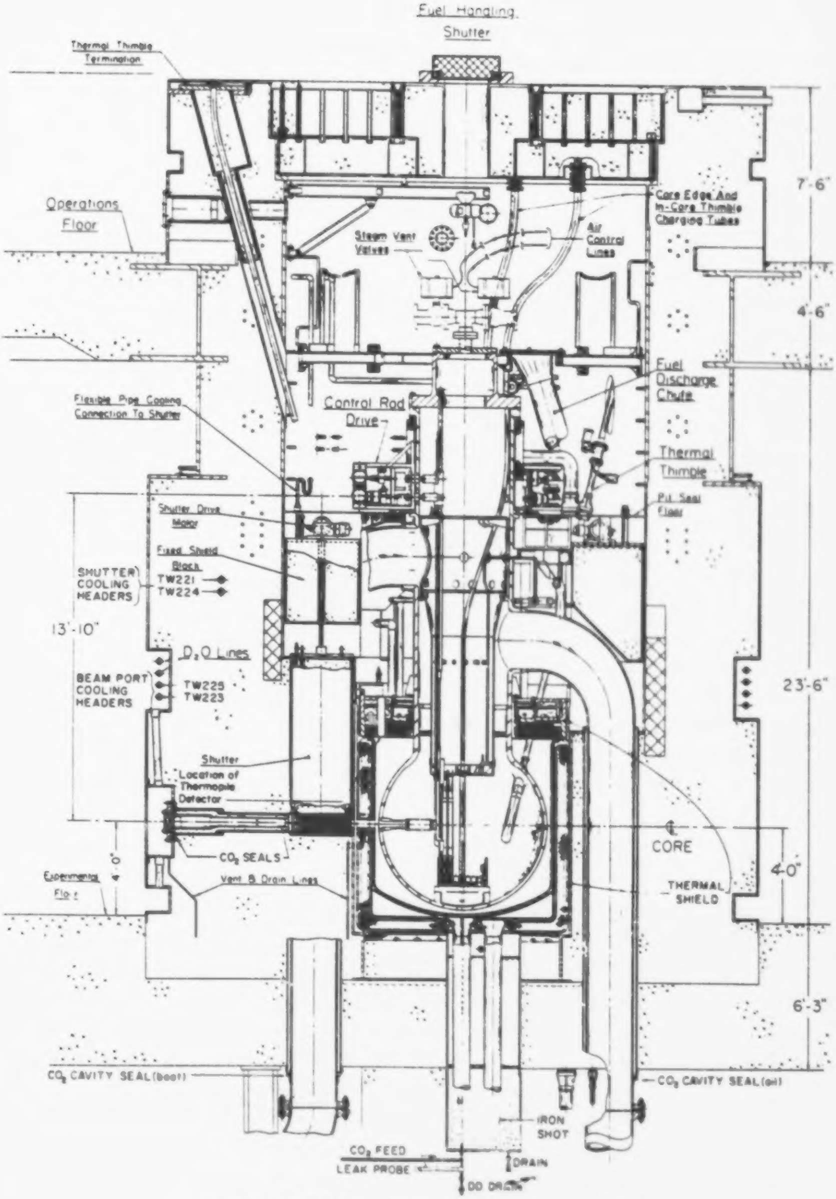

(18) Inner Fuel Plates, 23¾ Long, .050" Thick.
Fuel Alloy Core 22¾ Long .021" Thick.
Cladding .0145" Thick. Total U^{235} Content - 351 gms

(2) Outer Aluminum Plates .100" Thick.

57¼"
25½"
3½"
A
A
Core ℄
2.940
2.600
2.820
.187 Typ
.075 Min
2.236 Fuel Width
3.194
# 1 Water Channel .100 At ℄
2 .118
3 .108
4 .096
5
6
7
8
9
10
11
12
13
14
15
16 .096
17 .108
18 .118
19 .100
6" Rad

HFBR
FUEL ELEMENT
"KM"

Section A-A


Fuel Handling Shutter
Thermal Thimble Termination
Operations Floor
Steam Vent Valves
Core Edge And In-Core Thimble Charging Tubes
Air Control Lines
7'-6"
4'-6"
Fuel Discharge Chute
Flexible Pipe Cooling Connection To Shutter
Control Rod Drive
Thermal Thimble
Pit Seal Floor
Shutter Drive Motor
Fixed Shield Block
SHUTTER COOLING HEADERS
TW221
TW224
13'-10"
D₂O Lines
BEAM PORT COOLING HEADERS
TW225
TW223
23'-6"
Shutter
Location of Thermopile Detector
CO₂ SEALS
℄ CORE
4'-0"
4'-0"
Experimental Floor
Vent & Drain Lines
THERMAL SHIELD
6'-3"
CO₂ CAVITY SEAL (boot)
CO₂ CAVITY SEAL (oil)
IRON SHOT
CO₂ FEED
LEAK PROBE
DRAIN
DO DRAIN

HIGH FLUX BEAM REACTOR
ELEVATION

the secondary system (250 psig at the lowest elevation of the primary system versus 45 psig in the secondary), thus preventing light water from entering the primary system through any leak in the interface between the systems.

The required system pressure is determined by the need to prevent boiling at the hottest spot in a freshly fueled core. The boiling point of D_2O at atmospheric pressure is 214°F. With a hot spot surface temperature of 357°F, a pressure of 164 psig is required to suppress boiling. Helium gas at 200 psig is used as a cover gas in the reactor vessel to maintain the necessary pressure.

There are nine horizontal beam tube thimbles welded into the reactor vessel. The beam paths penetrate the thermal/biological shield. Eight of the beam tubes have 3.5-in. diameters and extend to within a few inches of the core. One of them points directly at the core to provide a fast neutron beam. A typical flux supplied to an experimental area is 4×10^9 neutrons/cm^2/s. The ninth horizontal tube serves as a "cold neutron" facility; the tube of the cold neutron facility is 1 ft in diameter and stops at 1 ft from the core. The cold neutron facility uses 1.4 l of liquid hydrogen as moderator, producing a beam of very low-energy neutrons.

Also projecting into the reactor are seven vertical tubes (thimbles) used for sample irradiations. The tubes provide three positions for irradiating samples in the heavy water moderator, two very near the core, and two in the core center. A typical cylindrical sample volume is 3 in. long by less than 1 in. in diameter.

The HFBR accident analyses focus on loss-of-coolant and loss-of-flow events. Mitigation of a potential loss-of-coolant accident (LOCA) depends heavily on an elevated tank of light water that is poisoned with cadmium nitrate (a neutron absorber). (The contractor's analyses of potential LOCAs and the phenomenon of flow reversal are discussed in greater detail below.) In loss-of-flow scenarios at HFBR, whether the reactor coolant system is depressurized or not, the coolant, which is normally pumped downward through the core, stagnates, and then reverses direction, as a result of natural convection. This flow reversal cannot be allowed to occur immediately after shutdown from normal 60-MW operation, because immediately after shutdown the rate of decay heating is too high for flow reversal to occur with sufficient speed to prevent fuel melting. Thus, one of the two major system upgrades associated with an increase in power level from 40 MW to 60 MW was the addition of battery-powered pony motors to the coolant pumps. These motors serve to assure at least 3 minutes of downflow cooling in the event of a power blackout and delay flow reversal until the decay heat has fallen sufficiently to allow flow reversal to occur without fuel damage. (The other upgrade was in the primary heat exchangers.)

The HFBR and its control room are enclosed in a confinement dome designed to withstand an internal pressure of 2 psi. In normal operation, a slight negative pressure is maintained within the dome to ensure that any air leakage is inward. Exhaust gases are filtered and discharged to the atmosphere from a high stack.

The future of the HFBR depends primarily on aging and on the continued need for the facility. The reactor is 23 years old and the original design was for a 25-year lifetime. Extensive studies of aging have been conducted. The DOE asserts that HFBR will continue to operate until a planned advanced neutron source, which could replace both HFBR and HFIR, is operational in the late 1990s [National Research Council, 1988].

RECENT OPERATING HISTORY

Each operating cycle at HFBR lasts approximately 1 month. The reactor operates for 24 days, after which 14 of the 28 fuel elements are replaced. The turnaround time for refueling, maintenance, and surveillance testing is typically 4 to 7 days. In most years there are 11 cycles scheduled, leaving 1 month free for more thorough inspections or modifications. One recent cycle was dedicated to replacing some secondary water piping, and only 10 cycles are scheduled for FY 1988.

A review of the 47 UORs from the past 5 years does not show any serious recurring problems. The two most common problems were associated with cooling of the Cold Neutron Facility and spurious events caused by unrelated instrumentation anomalies that resulted in accidental scrams. The next most common problems were difficulties with the personnel access doors, resulting in temporary breaks in the building confinement, and leaks in the secondary cooling piping. (This piping has now been replaced.) The remaining reports cover a variety of incidents, ranging from incoming power interruptions to administrative shutdowns. Two significant unusual occurrences during the past five years are described below:

1. In March 1986 the stator of the A primary pump motor failed because of unanticipated aging and caused a reactor shutdown. The following cycle was run at 40 MW, single loop operation. At the next

shutdown in April, the motor was replaced and operation resumed at 60 MW. The B primary-pump motor was replaced in 1987 as a precaution.

2. In December 1984 the main control rod M1 failed to fully insert following a scram. The rod stopped 4 inches short of the usual 30 inch insert position. The incident occurred during shutdown, with the reactor unfueled and depressurized. The cause was never determined. It is speculated that foreign material in the internal drive mechanism may have later cleared itself.

HFBR has had about four major plant modifications per year since start-up in 1965. Currently, there are 13 such modifications in the planning stage, including a system to provide remote monitoring of HFBR plant variables, remote control of shutdown cooling, and provision of a remote, alternative supply of light water for the poison water tank.

TECHNICAL ISSUES

60-MW Operation

Conclusion: The change in the operation of the HFBR from a power level of 40 MW to a power level of 60 MW has involved a significant change in safety philosophy.

At 40-MW operation, the HFBR safety analysis report indicates that no fuel melting would be expected to occur over a broad range of accidents [BNL, 1964a]. It states, for example, that "the system has been designed so that even a gross rupture of the primary vessel or primary coolant lines will not uncover the core and cause fuel melt." This statement is based on the design of the HFBR that involves a catch tank around the primary vessel that can prevent the core from being uncovered in the event of leakage, such as from the rupture of a beam tube. The reactor also has a flow reversal capability that allows core cooling to switch from forced downflow cooling to natural circulation in which the flow is upward through the core. Out-of-pile tests were conducted to establish the conditions for flow reversal. Based on the test results, fly wheels were incorporated on the primary pumps to ensure that, in the event of loss of power to the pumps, cooling would continue until flow reversal could be safely established after shutdown from 40 MW.

A reanalysis was made before operation at 60 MW and an addendum (dated April 1982) to the Final Safety Analysis Report was prepared and approved [BNL, 1982]. This addendum and the supporting documentation indicate that about 3 min. of downflow cooling must be maintained following shutdown from 60 MW. Pony motors were incorporated to provide greater assurance that 3 min. or more of downflow cooling would be provided. However, in the event of any system rupture greater than the equivalent of about a 1-in. diameter pipe, the loss of coolant is estimated to lead to pump trips and the loss of the necessary 3 min. of downflow cooling. Thus, unlike operation at 40 MW, such a break at 60 MW could lead to fuel melting. In addition, the failure of the driving mechanism for the "auxiliary" control rod that is used for control of the lower core region is expected to cause "some fuel melt" at 60 MW, whereas "no expected fuel melt" was predicted for 40-MW operation.

The approval of the change to 60-MW operation was based on a review of the potential consequences to public health by BNL staff. Dose estimates at the laboratory site boundary from fuel melting were less than the limits for commercial reactors found in 10 CFR 100.[1] While estimates of doses to individuals beyond the site boundary were low, the increased probability of fuel melting at 60 MW increases the potential for effects to individuals at the site. The procedures for abnormal conditions (such as, the response to loss-of-coolant accidents) can necessitate operator actions (e.g., manually opening valves) within the confinement building. Although reviews by BNL indicate that these actions can be accomplished during an accident [BNL, 1986a-b, 1987b, 1987f], the dose rates to operators were not estimated because of the presumed low probability of an event that might lead to high radiation exposures. The Brookhaven PRA, scheduled to begin in July 1988, should attempt to determine whether this presumption is valid. There are plans in progress to modify the reactor to allow for remote operation of the most important systems, but operator exposures for these planned operations are not clear either.

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[1] These dose estimates were made for a spectrum of accidents involving fuel damage, including a hypothetical design-basis accident in which the entire core was melted [BNL, 1979c]. In all cases, the estimated impacts on the public beyond the site boundary were slight. Exposures to visitors, laboratory scientists, and other persons at the site are typically not included in such calculations. However, in connection with its approval of 60-MW operation, DOE did request and review the contractor's dose estimates for exposures to the onsite population from such an accident.

Recommendation: A realistic assessment should be made of potential dose rates from fission products and exposure to operators during possible accidents at HFBR to ensure that adequate protection is provided to operations personnel and to other individuals at the site. This assessment should include a realistic evaluation of the requirements for evacuation and remote control of reactor shutdown and for establishment of long-term core cooling. While the timely accomplishment of portions of the assessment may involve deterministic analyses, the assessment should be integrated with the PRA.

Flow Reversal

> *Conclusion: The design and operating limits to ensure adequate cooling during flow reversal were established based on tests designed and conducted in 1963. The need to simulate the dynamic thermal-hydraulic behavior of the reactor in the test was not considered, although in response to committee questions a preliminary analysis of inertial effects has been completed that indicates that the 1963 tests were conservative.*

As discussed above, the HFBR design allows the coolant flow to reverse from forced downflow cooling to natural circulation with upflow through the core when the reactor is shut down from full power. The decay heat generated in the fuel plates must be adequately removed during this flow reversal process, or fuel melting may result. Prior to the initial start-up of the reactor, a series of out-of-core experiments were conducted with simulated fuel channels and a mockup of the flow reversal system. The tests indicated that core cooling during the flow reversal could be expected to occur without fuel damage [BNL, 1963].

During its review, the committee raised questions with BNL staff as to the adequacy of the test loop to simulate the reactor. In particular, the committee focused on dynamic thermal-hydraulic effects in the period during which the heated fuel plates experience a flow-stagnated condition, such as the time required for flow over the heated plates to decelerate and reverse. BNL believes that the temperature limits based on the original test results are conservative even when such dynamic effects are taken into consideration [BNL, 1988m]. Since dynamic effects were not considered in the original design of the experiments, it is important that a more careful and thorough review be undertaken to confirm that the limits are indeed conservative. Analysis of the flow-reversal phenomenon at HFBR should be conducted using modern numerical simulation techniques.

Recommendation: A preliminary analysis of flow reversal effects at HFBR previously provided to the committee should be completed and reviewed by the Brookhaven Reactor Safety Committee, by DOE, and by the DOE Advisory Committee on Nuclear Facility Safety. This bounding analysis should be supplemented by a numerical simulation using modern reactor codes to determine whether the tests conservatively model the heat-up processes of the reactor fuel under flow reversal conditions and whether the limits derived from the experimental tests are valid.

Control Room Staffing

> *Conclusion: The current method of operation allows the control room to be manned by only one person during all or any part of a shift.*

It is current practice at HFBR to have two operators and one supervisor on shift. (The supervisor is a qualified operator.) The main duties of operating the plant are performed in or near the control room, but various actions and checks are required either routinely or for other reasons in various parts of the plant. Under the current procedures both the supervisor and one of the operators could be away from the control room simultaneously. In the committee's judgment, it is not good practice to allow the control room of an operating reactor to be manned by only one qualified operator. There should be at least two qualified operating personnel in the control room at all times during reactor operation and during refueling to respond to occurrences that could affect the safety of the plant.

Recommendations: The technical specifications for the HFBR should be revised to require that the control room be manned at all times by no fewer than two qualified operating personnel while the plant is operating or being refueled. DOE should develop a consistent policy for control room staffing for all of the test and research reactors.

Light-Water Ingress

> *Conclusion: The addition of light water into the HFBR core region can cause substantial reactivity increases. The safety*

of the plant depends upon continued vigilance and training to avoid a reactivity-induced excursion from the addition of light water. Operators may be exposed to a high-radiation environment in using the existing emergency system for addition of light-water coolant.

Light-Water (H_2O) Injection into the Primary Heavy-Water (D_2O) Cooling System

As described in the FSAR, the addition of light water to the HFBR core can substantially increase reactivity. The addition of light water would also reduce the effectiveness of the reflector control rods. It is possible under these circumstances for the light-water flooded core to become super-critical even with all the control rods inserted. This means that power levels could increase very rapidly.

To avoid a potential reactivity excursion owing to the addition of light water, the contractor controls all sources of light water and conducts reviews to ensure that any significant injection of light water is avoided. There is only one direct connection in HFBR between a light-water system and the primary heavy-water system. This connection is through a tank filled with a solution of cadmium nitrate. The system is designed so that the solution of cadmium nitrate (a neutron "poison") can be added if needed as a backup method to assure reactor shutdown and to keep the core covered in the event of a loss of heavy water coolant.

If light water were needed to keep the core cooled, the system would be actuated manually by aligning three valves in the piping leading to the tank full of cadmium nitrate. After the cadmium nitrate tank is empty, additional light water can only be added by operation of an additional spring-loaded valve. The spring-loaded valve was designed to close automatically when released so that it cannot be inadvertently left in the open position. The operating procedures specify that light water is to be added only to make up for losses from boiling. The practice of permitting only limited makeup of coolant is to ensure that the cadmium nitrate neutron poison is not flushed out of the core by subsequent addition of light water; poisoned coolant must remain in the core to prevent re-criticality. The operator must make the appropriate adjustment of valves during an accident within the confinement—that is, in a potentially hazardous environment.

Although the committee was informed that the operator at the spring-loaded valve cannot directly read the water level in the reactor vessel—an essential measurement if light-water addition is to be limited to the replacement of losses caused by boiling—remote water-level readings are being installed this year (1988). The concentration of cadmium nitrate in the tank is checked annually in accordance with the reactor technical specifications, and no problems in maintaining proper concentrations have been found. The overall system and other sources of possible light-water addition were reviewed by BNL in August 1986, and the possibility of light-water flooding of the core without neutron poisoning was estimated by the contractor to be acceptably small [BNL, 1986b]. The contractor's PRA should aim to determine whether this conclusion is valid as well. The light-water addition system is sufficiently dependent on proper operator action that special care must be taken.

Light-Water Injection into the In-Core Thimbles

HFBR has in-core thimbles that are used to irradiate materials samples in the high neutron flux regions of the reactor. These thimbles are cooled internally by the circulation of heavy water through an "experimental cooling system" that is separate from the primary heavy-water system. The heat exchanger for this system is cooled by light water at a pressure higher than the pressure of the heavy water in the experimental cooling system. The committee is concerned about the possible effects of light-water leakage into the experimental cooling system and subsequent injection into the in-core thimbles, causing a positive reactivity insertion. This question has not been analyzed in the FSAR. Preliminary analysis by BNL in response to our inquiry suggests that any such leakage would not be a problem [BNL, 1988k], but the analysis needs to be formalized and properly reviewed.

Recommendation: Continual vigilance is necessary to ensure that light-water (H_2O) additions to the heavy water (D_2O) cooling system do not inadvertently occur. Because operator actions are involved, this vigilance should include special training to explain not only the procedures but also the basis for the requirements, and thereby to motivate the operators to take special care to ensure that the neutron absorbing cadmium nitrate solution is always added and remains present during any light-water additions. Training and analysis should give due *consideration to unexpected conditions, for example, possible stratification or over-dilution of the cadmium nitrate solution. Planned improvements should be implemented to provide remote reading of the water level in the core at stations where light water can be added to the reactor coolant system. A realistic assessment should be made of potential dose rates and operator exposures during manual operation of the light-water addition system.*

The contractor's preliminary analysis of water injection into the in-core thimbles should be formalized, including an analysis of transients and consequences. This analysis should be reviewed and added to the Final Safety Analysis Report.

Beam Tube Embrittlement

> *Conclusion: Beam tubes in the HFBR are being embrittled by prolonged exposure to neutron irradiation. Although the problems encountered at neutron scattering facilities abroad as a result of beam tube embrittlement have not yet arisen at the HFBR, tube embrittlement may limit the useful life of the reactor.*

The HFBR has beam tubes that channel neutrons outside the reactor vessel for use in neutron scattering experiments. The beam tubes are integral parts of the reactor vessel. Safe operation of the reactor thus requires maintenance of the integrity of these tubes.

The beam tubes are positioned close to the reactor core and receive very high exposures to neutron irradiation. Neutron irradiation embrittles metals; indeed, several different types of problems have arisen at high flux reactors in France because of embrittlement.

When the HFBR was designed and constructed, it was assumed that vessel embrittlement would occur progressively through atomic displacement. To monitor the progressive embrittlement of the vessel over the life of the reactor, surveillance specimens of 6061-T6 aluminum alloy (of which the beam tubes were made) were suspended in regions of high neutron irradiation. As expected, examination of these specimens has shown reductions in metal ductility as a result of prolonged irradiation. However, the reduction has been caused by the transmutation of aluminum atoms to silicon atoms rather than as a result of atomic displacement [BNL, 1988e, 1988i].

Certain French reactors have experienced complete loss of beam tube ductility through prolonged irradiation. However, the aluminum alloys used in these reactors contained significant amounts of magnesium. The loss of ductility was the result of the transmutation of aluminum to silicon, which led to the prompt precipitation of intermetallic compounds of magnesium silicide. When fabricated, the 6061-T6 alloy used in the HFBR contained little magnesium, and the magnesium that was there was largely reacted with silicon during the fabrication process. Consequently, the HFBR alloy is not as susceptible to rapid embrittlement as alloys rich in magnesium, like the ones used in the French reactors. Nor does it appear that the HFBR alloy is as susceptible to stress corrosion cracking or pitting corrosion as the French alloys.

Nevertheless, as irradiation progresses, more and more silicon is produced in the tips of the HFBR beam tubes. Because silicon has a limited equilibrium solubility in aluminum, the potential for precipitation of brittle silicon in the alloy increases with continued irradiation. The equilibrium solubility of silicon in aluminum is only about 1 percent. Silicon concentrations of nearly 8 percent have been observed in some surveillance specimens at HFBR. There may be kinetic barriers to the nucleation of silicon precipitates that explains why precipitation has yet to occur. However, as the level of supersaturation increases, it is likely that these kinetic barriers would be less effective in preventing precipitation. Indeed, at some point, sudden precipitation of silicon could occur spontaneously. Silicon concentrations as high as 14 atom percent have been projected for the HFBR over the next 10 full-power years.

The contractor and DOE are aware of the progressive embrittlement of the HFBR beam tubes, and the threat that it poses. They have solicited external review and advice on the potential impact on the integrity of the HFBR coolant system. Unfortunately, little is known about the behavior under irradiation of 6061-T6 alloy at neutron fluences greater than those already experienced at the tips of the HFBR beam tubes. The expert reviews that were undertaken concluded that continued monitoring and study of tube embrittlement are needed.

Because the HFBR is cooled with heavy water, some tritium is produced during normal operation. The contractor believes that cracking or perforation of the beam tubes would be detected by tritium leakage before catastrophic rupture could occur [BNL, 1987d]. But the nature of tritium leakage from a defective beam tube has not been defined.

Quite clearly, there is a need to monitor closely the loss of ductility of the HFBR beam tubes. With increasing exposure, the continued integrity of these tubes will be more suspect. If the tubes cannot be replaced at reasonable cost, embrittlement will eventually limit the useful life of the reactor. Meanwhile, the contribution of beam tube embrittlement and potential tube rupture to the risk of accidents at the HFBR needs to be fully assessed, along with the risk of a beam tube rupture exacerbating an accident initiated by some other means.

Recommendation: The committee endorses continued monitoring and study of beam tube embrittlement at HFBR. The contractor should pay careful attention to uncertainties in beam tube rupture in the PRA currently under way at the HFBR. DOE should explore cost-effective means for replacing the HFBR beam tubes.

High Flux
Isotope Reactor

105

High Flux Isotope Reactor (HFIR)

The High Flux Isotope Reactor (HFIR) is located in eastern Tennessee at the Oak Ridge National Laboratory (ORNL). It is operated on behalf of the DOE Office of Energy Research by Martin Marietta Energy Systems, Inc. The reactor reached its design power of 100 MW in 1966. Its primary mission is the production of transuranic radioisotopes. The types and amounts of radioisotopes produced at HFIR in 1986 are shown in Table 2. HFIR provides significant materials irradiation capabilities, as well as facilities for neutron scattering experiments. The operation staff consists of 88 people, and the FY 1987 budget was $11.4 million. The FY 1988 budget request was $19.6 million.

HFIR is a flux-trap reactor that is designed to provide especially high fluxes of neutrons that have thermal energies. Thermal neutrons at high fluxes are needed to produce transuranic isotopes efficiently. The reactor uses light water as coolant and beryllium as reflector. The core is designed as a series of concentric cylinders approximately 30 in. high with an active fuel region that is 20 in. long. The central shaft, 5 in. in diameter, comprises the flux trap. When a target is inserted in the core, the average flux in this area is about 5×10^{15} neutrons/cm^2/s, and about one-half of the flux is thermal.

The core consists of two annuli of fuel plates. There are 171 plates in the inner annulus and 369 in the outer. The plates consist of a

TABLE 2 Isotopes Produced at HFIR in FY 1986

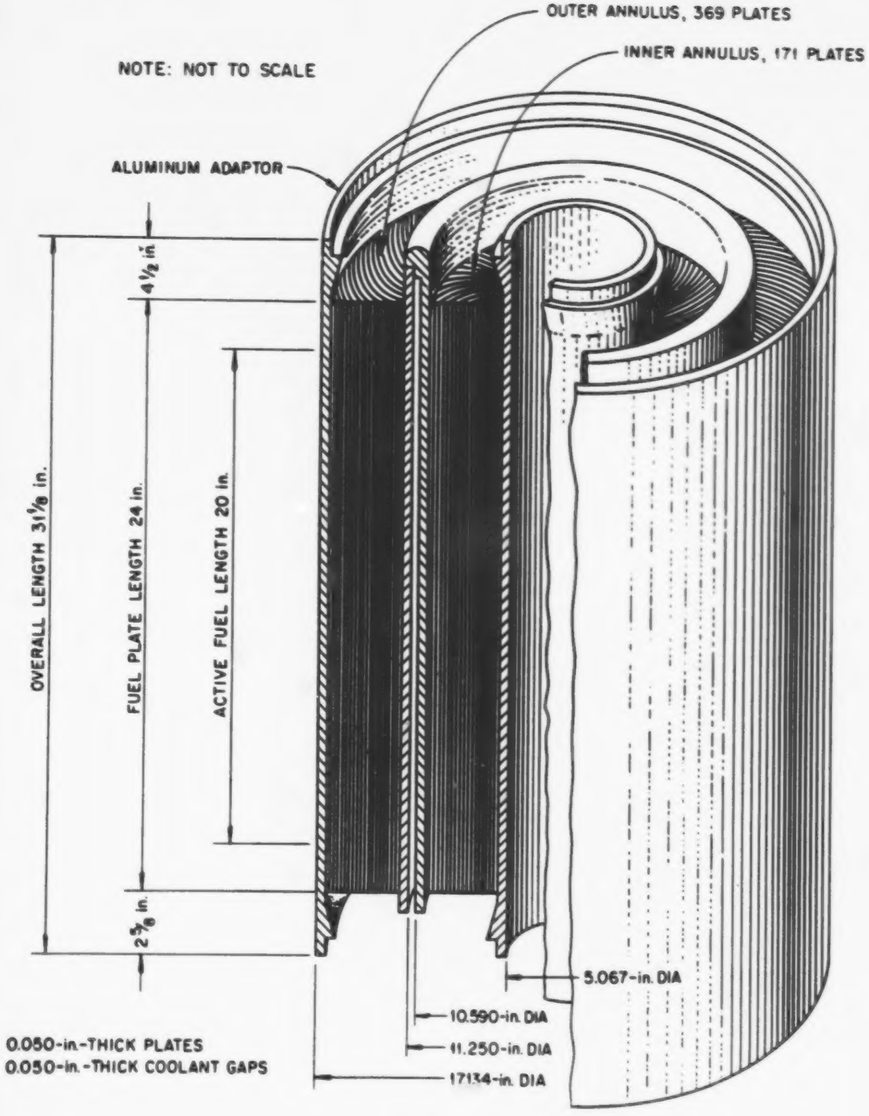
Element (arranged by weight)	Half-life	Mass Produced	Radioactivity (millicuries)
Phosphorus-33	25.4 d	0.005 mg	$8.1\text{x}10^{2}$
Calcium-41	$1.04\text{x}10^{5}$ y	5.894 g	$5.0\text{x}10^{2}$
Iron-55	2.7 y	0.410 mg	$1.0\text{x}10^{3}$
Cobalt-60	5.27 y	226.2 g	$2.6\text{x}10^{8}$
Nickel-63	100. y	2.6 g	$1.5\text{x}10^{5}$
Selenium-75	119.8 d	0.646 mg	$9.4\text{x}10^{3}$
Strontium-85	65.2 d	0.032 mg	$7.5\text{x}10^{2}$
Zirconium-95	64. d	0.003 mg	$5.6\text{x}10^{1}$
Tin-119m	293. d	0.959 mg	$3.6\text{x}10^{3}$
Barium-135m	28.7 h	0.014 mg	$1.1\text{x}10^{4}$
Gadolinium-153	242.4 d	212.0 mg	$7.5\text{x}10^{5}$
Ytterbium-169	32. d	0.207 mg	$5.0\text{x}10^{3}$
Iridium-192	74. d	53.6 g	$4.9\text{x}10^{8}$
Curium-248	$3.40\text{x}10^{5}$ y	150. mg	$6.3\text{x}10^{-1}$
Berkelium-249	320.d	50. mg	$8.2\text{x}10^{4}$
Californium-252	2.64 y*	500. mg	$2.7\text{x}10^{6}$
Einsteinium-253	20.4 d	2. mg	$5.1\text{x}10^{1}$
Einsteinium-254	275.7 d	4. g	$7.5\text{x}10^{3}$
Fermium-257	100.5 d	1. pg	$5.1\text{x}10^{-6}$

*Cf-252 has a neutron output from spontaneous fission of 2.3×10^{12} neutrons/s/g.

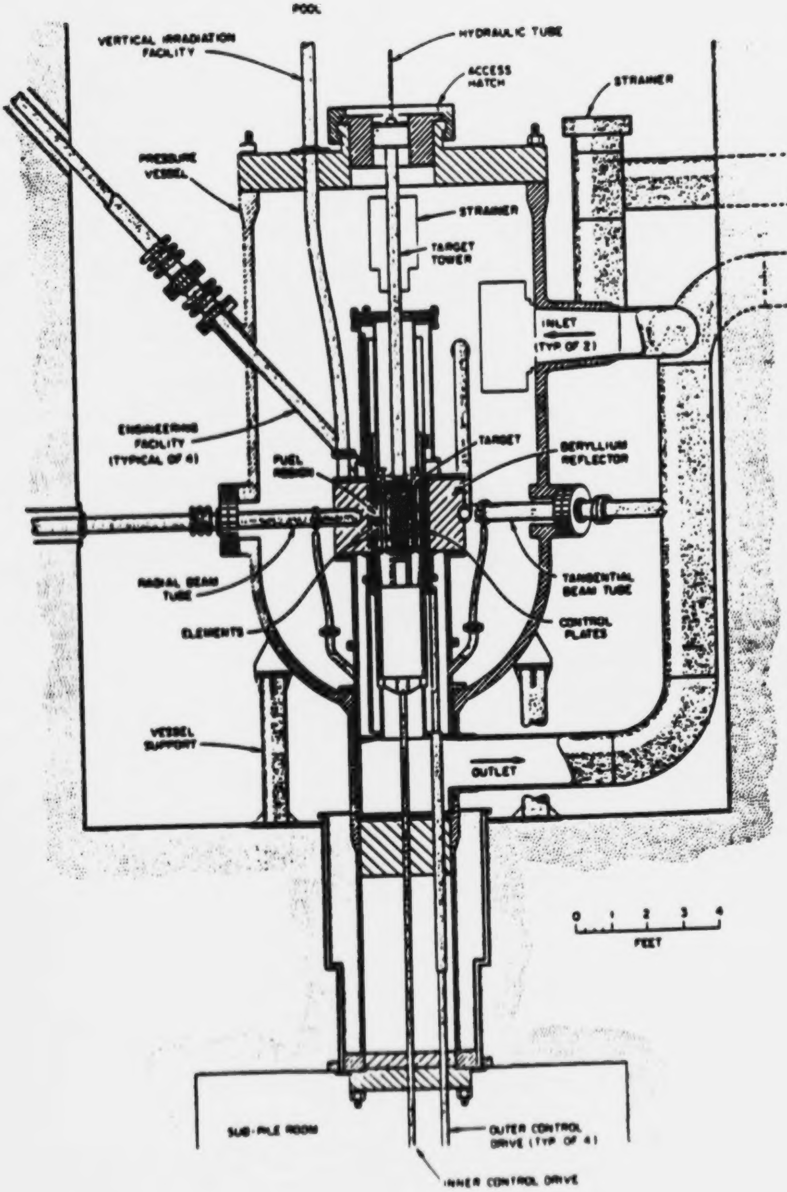
cermet fuel made of U_3O_8 and aluminum, with aluminum cladding. The uranium is enriched to 93 percent U-235. A fresh core contains about 9.4 kg of U-235 and 2.8 gm of Boron-10.

Immediately outside the fuel annuli are two neutron-absorbing control cylinders, each about 0.25 in. thick. The cylinders are each three times the height of the core, and have different absorbers at different altitudes. The net reactivity is altered by moving the control cylinders up or down. The absorbers are tantalum and europium oxide (Eu_2O_3).

Outside the control cylinders is the beryllium reflector, constructed in a series of annular cylinders that total to about 1 ft in thickness. The four external beam tubes protrude into the inner half of the reflector cylinders to pick up neutrons. The 20 vertical experimental irradiation facilities are in the outer portion of the reflector; flux there is about 1×10^{15} neutons/cm^2/s. The outer periphery of the reflector also includes four tubes to channel neutrons from


OUTER ANNULUS, 369 PLATES
INNER ANNULUS, 171 PLATES
NOTE: NOT TO SCALE
ALUMINUM ADAPTOR
4½ in.
OVERALL LENGTH 31⅛ in.
FUEL PLATE LENGTH 24 in.
ACTIVE FUEL LENGTH 20 in.
2⅝ in.
5.067-in. DIA
10.590-in. DIA
11.250-in. DIA
17134-in. DIA
0.050-in.-THICK PLATES
0.050-in.-THICK COOLANT GAPS

HFIR Fuel Element


POOL
VERTICAL IRRADIATION FACILITY
HYDRAULIC TUBE
ACCESS HATCH
STRAINER
PRESSURE VESSEL
STRAINER
TARGET TOWER
INLET
(TYP OF 2)
ENGINEERING FACILITY (TYPICAL OF 4)
FUEL REGION
TARGET
BERYLLIUM REFLECTOR
RADIAL BEAM TUBE
TANGENTIAL BEAM TUBE
ELEMENTS
CONTROL PLATES
VESSEL SUPPORT
OUTLET
0 1 2 3 4
FEET
SUB-PILE ROOM
OUTER CONTROL DRIVE (TYP OF 4)
INNER CONTROL DRIVE

Cross Section of HFIR Reactor

the reflector exterior to an experiment room located above the main beam room.

The pressure vessel is eight feet in diameter, and is made of 3-in. stainless-clad carbon steel. It sits in a reactor pool, which is 18 ft in diameter and filled with water. The top of the reactor pressure vessel is 17 ft below the surface of the water. The pool contains the reactor vessel, provides biological shielding, serves as a heat sink and a source of emergency makeup water, and offers access to the storage pool. The reactor pool in which the HFIR is submerged is a noteworthy safety feature. In many of the hypothesized accidents that might occur at HFIR, any fission products released from the fuel would have to pass through the 17 ft of water in the pool. It is well established that as long as the water in the pool remains below the boiling point, it could be effective in scrubbing cesium and iodine, but not noble gases, from gases sparging through it.

Three vertical centrifugal pumps propel the light water coolant through the primary loop at about 17,000 gpm. The vessel inlet pressure is 750 psi. The coolant enters through two pipes high on the vessel, and flows downward through the core, entering at 120°F and exiting at 160°F. It leaves the bottom of the vessel through one large pipe, which feeds three of four parallel heat exchangers; the fourth is kept on standby. The secondary loop subsequently dumps the heat into three of four conventional cooling towers.

The neutron scattering facilities consist of eight spectrometers fed by the four beam tubes. Typical beam tube flux for the monochromatic beam at the sampling position is 10^7 neutrons/cm^2/s. There is also a small-angle scattering facility operating at much lower flux.

The reactor confinement normally operates under a small negative pressure. The interior air is exhausted to the atmosphere through banks of filters and then through a 250-ft stack.

HFIR has not operated since November 1986. ORNL shut the reactor down when it was discovered that the pressure vessel was being embrittled by exposure to neutrons at an unexpectedly rapid rate. Reactor operations have remained suspended as investigations have been conducted into the embrittlement problem and the causes of the delay in its discovery. The contractor has also labored to respond to various recommendations for improving other aspects of plant operations while the reactor has been shut down.

RECENT OPERATING HISTORY

In the 5 years preceding shutdown in November 1986, HFIR averaged an annual availability factor of 84 percent, with only about one week of unplanned outage per year. When operating at 100 MW, the nominal fuel cycle was 22.5 days.

In a typical year of operation 25 percent of the unusual occurrences reported to DOE resulted from a loss of AC power to the reactor, usually caused by lightning. Another 25 percent was due to trips resulting from extremely low trip thresholds during tests of the safety subsystems.

Recent operating events prior to the 1986 shutdown included the following:

- In January 1984, with the reactor at about 40 percent power, the seat light for one of the safety rods actuated in the control room, and the operators manually scrammed the reactor. When the reactor was scrammed, the safety rod in question failed by about 3/4 in. to insert completely. A subsequent investigation indicated that the lower section of the safety rod had fallen off. This section, called the piston section, is normally screwed onto the assembly and staked by peening in three places. In this case all three stakes had worn through, and vibration caused the piston to unscrew itself.
- In January 1986, tantalum was detected in the demineralizer effluent. Investigation revealed a failure of the cladding over the tantalum on three control plates of the inner control cylinder. The failure was traced to inconsistent epoxy bonding during fabrication in 1976. Originally there were five such plates that were found to be "nonconforming" but were considered serviceable. Three of them were leaking tantalum by 1986. Following the incident, the three damaged control plates were replaced with spare plates, one of which was one of the original five defective plates. After shutdown in November 1986, inspection of this plate showed a 1-in. diameter blister in the cladding of the europium control material. An investigation by the contractor concluded that "failure appears imminent, so the cylinder will not be used in the future," but in any event "the only result would have been a slight contamination of the primary coolant system, which could have been cleared up with no operational problems."

In the 20 years of operation preceding the current shutdown, there were about 180 plant modifications. All but two of the modifications were regarded as updates of existing systems or components, and, as such, were interpreted as not requiring DOE review or approval. This included phase one of the HFIR Irradiation Facility Improvement (HIFI) project, which involved redesign of the target tower assembly and associated reactor components. Other significant plant modifications since startup included improved bearings and bearing mounts for the control plates, coolant strainers with longer service life and varying mesh sizes, and the upgrading of electronics with integrated circuits.

Following the November 1986 shutdown there have been about 10 modifications proposed, all of which were submitted to DOE for review and approval. According to the contractor, these modifications have now been completed. Two of them involved installation of a seismic scram capability and the strengthening of the primary system and reactor pool against earthquakes. The remainder were associated with monitoring and maintaining a new lower limit for operating pressure and temperature.

TECHNICAL ISSUES

Management of the High Flux Isotope Reactor

Conclusion: DOE and the contractor have determined that management deficiencies by both organizations were the cause of serious breakdowns in the safe operation of HFIR. Restructuring of management is now under way.

When the HFIR was constructed, it was recognized that over time the pressure vessel steel would be embrittled by exposure to neutron irradiation. Surveillance specimens of various steels used in the construction of the pressure vessel were suspended in the reactor so that the rate of embrittlement could be monitored. As discussed in greater detail below, in 1986 it was discovered that specimens withdrawn for analysis in 1983 had not in fact been analyzed. When these specimens were analyzed, it was found that the reactor had been operating for several years at temperatures that were outside the technical specifications—that is, outside limits that had been established for safety reasons.

Discovery of the vessel embrittlement problem prompted closer examination by DOE and the contractor of the administrative and management controls in place for operating the reactor. A large number of reviews, audits and appraisals were conducted at the instigation of both the contractor and DOE. A particularly critical review of contractor management was conducted by DOE's ES&H organization [DOE, 1987c]. The ES&H review called attention to eight major deficiencies:

- Insufficient upper level management emphasis on safety.
- Insufficient resources expended for safe operation of the reactor.
- Inadequate self-appraisals and safety reviews.
- Inadequate documentation and analysis of safety-related data and events.
- Inadequate operator training.
- Lack of adequate organizational controls, standards, and procedures.
- Lack of adequate control of nonoperational and maintenance activities that could affect reactor operation.
- Lack of effective functional quality assurance program.

Taken together, these eight deficiencies represented a severe indictment of operations at HFIR.

During its visit to the reactor, the committee found that the contractor was attempting to respond fully to the criticisms. The considerable expertise of ORNL had been applied to reviews of the reactor and to the development of probabilistic risk and severe accident analyses. In addition, upper management of reactor operations had been changed. Nevertheless, the committee concluded that some of the problems identified in prior reviews appeared not to have been effectively resolved. It was apparent, for example, that upper management was largely unfamiliar with the actual condition of systems within the plant. Leaking valves and jury-rigged plumbing indicated that maintenance of reactor facilities still lagged behind minimum expectations. The contractor's "matrix" organization for conducting maintenance (see below) was still in force. There were indications of inadequate housekeeping.

Breakdowns of the magnitude of those that were found at HFIR demonstrate failures by the local operations office as well as the contractor. An important function of DOE's operations offices is to detect and correct breakdowns of contractor management and administration. Careful oversight of the contractor is the principal mechanism available to detect lapses in management and to alert those with ultimate responsibility for safe operation to the need for corrective action.

Subsequent to the committee's visit, the contractor and the department have made further management changes. These changes are intended to increase top management involvement by the contractor and the operations office and to further increase DOE staff resources. Assessments of the management system by experts with commercial power experience and by departmental headquarters are to be completed prior to restart.

Recommendation: DOE must ensure that the steps taken to correct deficiencies previously identified at HFIR have been effectively implemented both by the contractor and by the local DOE operations office.

Vessel Embrittlement

> *Conclusion: The HFIR contractor has formulated an operating strategy that is thought to avoid the hazards posed by pressure vessel embrittlement. The strategy is the product of considerable, external expertise that the contractor and DOE have brought to bear on the problem of embrittlement. However, insufficient attention has been paid to the consequences of vessel failure and more careful evaluation of vessel failure should be included in the PRA.*

The HFIR has a pressure vessel consisting of carbon steel clad with stainless steel. At the time HFIR was designed, it was well known that the carbon steel base metal of the pressure vessel would undergo microstructural changes when subjected to irradiation by high energy neutrons. One effect of these changes results in a reduction in the ductility of the steel. All carbon steel pressure vessels undergo a change from ductile fracture behavior to brittle fracture behavior as the temperature of the metal decreases. The transition temperature between ductile and brittle behavior is called the nil-ductility transition temperature (NDT). Under prolonged energetic neutron irradiation, the NDT of the carbon steel that constitutes the major part of the pressure vessel increases.

The base metal of the pressure vessel, beam tube nozzles, and vessel welds of HFIR are made of different metal alloys with different NDTs. At start-up, the highest initial NDT of any of the metal used in the HFIR pressure vessel was 0°F. All of the evidence available at the time the reactor was designed indicated that any change in the

NDT of the vessel material exposed to the highest flux in the reactor would be negligible, even after 20 years of full power operation.

However, because radiation-induced embrittlement of steel was incompletely understood, a conservative technical specification was imposed to keep the plant from being operated while pressurized at temperatures below 70°F. This technical specification was expected to ensure that throughout the life of the vessel the NDT of the steel would never exceed the design criterion, which was that temperatures of the vessel should always remain 60°F higher than the NDT (T > NDT + 60°F). This would serve to ensure that the vessel would not fail as a result of cracking.

To monitor embrittlement, steel specimens (termed "coupons") were suspended in the reactor coolant adjacent to parts of the reactor vessel shell and beam tube nozzles. These specimens were located in places that would enable surveillance of the range of vessel materials at points where the materials were subject to low and high fluxes of energetic neutrons. The plan was for specimens of the various materials to be withdrawn periodically and tested to ensure that the original projections concerning the effects of irradiation on the ductile-to-brittle transition of the HFIR pressure vessel steels remained valid.

The practice of withdrawing and testing specimens of nozzle material was followed for a number of years. Data from the specimens indicated nominal shifts in the NDT. However, specimens of shell material—from which the reactor pressure vessel shell was constructed—including specimens at the location in the vessel of the highest neutron flux, were not withdrawn and tested at all prior to 1983. In 1986, during a post-Chernobyl review, it was discovered that none of the specimens withdrawn from the reactor 3 years previously had been tested [DOE, 1987a]. The specimens withdrawn in 1983 included, for the first time, specimens of both nozzle and shell materials. When the 1983 specimens of high flux shell material were tested in 1986, they showed estimated NDTs for the vessel wall of 55°F. If during this period the reactor was pressurized at 70°F—and the contractor cannot establish that it was not—this would mean that it was operated in violation of technical specifications (NDT + 60°F = 115°F). Additional samples of the same material pulled in 1986 showed an NDT of 75°F for the vessel wall. The samples thus show that the reactor may have been operated not only in violation of technical specifications, but also under circumstances in which portions of the reactor vessel were susceptible to brittle fracture.

The cause of the relatively rapid change in the NDT of the HFIR pressure vessel steels is not well understood. It has been hypothesized that slow irradiation by the unusual neutron spectrum from HFIR produces a rate of irradiation-induced embrittlement not heretofore known. Information on the embrittlement rate is scant because of the limited number of surveillance specimens and the type of specimens used at HFIR. Indeed, estimation of the rate of embrittlement of the HFIR pressure vessel is very uncertain because there are only two data points for vessel shell material exposed to the highest flux.

There are other uncertainties. The HFIR pressure vessel is a welded structure, and it is often found that weld materials and the heat-affected zones surrounding welds are more susceptible to embrittlement. Unfortunately, weld materials were not included among the surveillance specimens in HFIR. However, inferential data from other types of irradiation tests suggest that welds at HFIR are no more affected by irradiation than the bulk metal.

In the face of these uncertainties, the HFIR contractor has developed a strategy for continued operation [MMES, 1987l]. The contractor has proposed to operate the reactor at the same inlet temperature of 120°F but at a reduced power of 85 MW and a reduced pressure of 485 psi. A heating system will be installed so that pressurization of the reactor only occurs when temperatures in the pool are above 90°F. The contractor is also proposing to conduct a hydrostatic test of the pressure vessel annually at 900 psi and 85°F. The test is intended to generate ratios of applied stress to fracture toughness of the metal that exceed any ratio of stress to fracture toughness that could develop during the next year of operation. In other words, if the vessel were to be susceptible to failure before the next hydrostatic test, the contractor asserts that the failure would occur during the test rather than during operations [MMES, 1988n]. The first of these hydrostatic tests was conducted last year. Although some unexplained acoustic signals were received during the test, the external reviewers and the contractor concluded that the test did not yield evidence of vessel fragility [BNL,1988; MMES, 1988n]. (The DOE Advisory Committee has established a subcommittee to evaluate the degree to which the hydrostatic test provides assurance that the vessel will not fail during operation.)

The proposed basis for continued operation is not consistent with the original criterion for pressure vessel integrity (NDT plus 60°F). Furthermore, the new operating regime results in a margin of no more than 10°F between the estimated NDT of HFIR pressure vessel steel and the new lower temperature limit for pressurized reactor operation, even though the estimated margin of error for mean NDT is ± 10°F. Thus, the new strategy does not provide any assured margin between the NDT and the temperature limit for pressurized operation, as specified in the original technical specifications. Any margin that does exist will be progressively reduced as the vessel is exposed to further irradiation. The rate of embrittlement in HFIR is not known; the contractor believes that embrittlement of the vessel is subject to saturation [MMES, 1988n], and that the rate of embrittlement is, at worst, linear, increasing by no more than 4° to 5° per year. Resumption of operation will require revised, well-justified technical specifications for pressurized operation.

The contractor and DOE have solicited extensive expert opinion concerning the problem of vessel embrittlement, and the proposed operating strategy has been approved by skilled and knowledgeable outside reviewers. The reviews appear to have been focused almost exclusively on the question of the likelihood of vessel failure under the proposed operating regime, given uncertainties regarding the physical processes involved in irradiation-induced embrittlement at HFIR [BNL, 1987; DOE, 1987b; MMES, 1987l]. In the committee's view, however, the risk of vessel failure must be judged not only in terms of the probability of occurrence, but also in terms of potential consequences. The only review of the consequences of vessel failure was conducted by the contractor, and it did not constitute an in-depth evaluation of all aspects of the problem [MMES, 1987l]. Rather, the analysis consisted primarily of a recapitulation of an earlier analysis of vessel failure that was conducted in conjunction with the HFIR accident analysis report written more than 20 years ago.

In essence, the contractor hypothesizes that sudden depressurization of the HFIR vessel would be no worse than the "maximum credible accident" analyzed in 1967 [Union Carbide Corp., 1967]. That is, rapid depressurization would induce fuel melting of 50 percent of the core, with attendant releases of a fraction of the radionuclide inventory. As a result of recent experiments conducted with fuels similar to those used in HFIR, however, the releases of radionuclides postulated in 1967 may not be conservative. Indeed, the contractor's own Reactor Review and Audit Committee raised questions in 1987 about the assumptions regarding releases contained in the original HFIR safety analyses [MMES, 1987a]. The committee concludes, therefore, that there has been insufficient reanalysis of possible accident consequences of vessel failure.

The Department and the contractor have recently conducted a Level 1 PRA of HFIR [Pickard, Lowe, and Garrick, Inc., 1988]. While accidents involving or initiated by pressure vessel failure were included in the study, the committee found these analyses unconvincing. It is not clear, for example, whether the conditional probabilities for vessel failure estimated by the PRA contractor take into account embrittlement of the reactor vessel. Moreover, the potential for vessel failure to exacerbate accidents initiated by other causes was not carefully examined. Both of these matters should be more carefully addressed. The committee recognizes, however, that the treatment of vessel failure in the HFIR PRA is uniquely challenging because of the absence of even generic data on the probability of vessel failure, because of technical uncertainties concerning the nature of the embrittlement processes at HFIR, and because of deficiencies in the understanding of radionuclide release once the HFIR vessel has failed.

Recommendation: The contractor should reanalyze the consequences of vessel failure in light of modern knowledge of radionuclide releases and modern methods of accident analysis. Additional expert review of the treatment of vessel failure in the HFIR PRA needs to be undertaken, incorporating known uncertainties concerning the probability and consequences of vessel failure.

Maintenance

> *Conclusion: Maintenance work at HFIR has been conducted by employees assigned to a central laboratory maintenance organization who are not under the supervisory control of the reactor manager.*

A number of staff changes have taken place at ORNL to strengthen the operating staff of HFIR and the safety review of the reactors. These have included assignment of new personnel to positions of responsibility, and enlargement of the operating staff in areas where experience has indicated it to be too thin in number and assured competence. These were necessary changes. However, there is one area affecting safety that continues to concern the committee: maintenance.

Maintenance in all areas at Oak Ridge is conducted on a "matrix" basis. When a maintenance activity is required that necessitates work by members of a particular craft, a work order is issued to obtain services from a central maintenance group. Thus, for example, the same group of pipe fitters, is drawn on for work at HFIR as is used for work in other, nonreactor areas of the laboratory. The work is supervised by the laboratory's central maintenance group. This means that the director of the HFIR is not in direct charge of all of the individuals who do work on the reactor. Such a practice, unless it is carefully controlled, can undercut the fundamental responsibility for safety, which attaches to the director of the reactor. There were indications in a January 1988 DOE review of quality assurance practices at HFIR that problems continue to occur in this area [DOE, 1988b]. In response to the DOE review, ORNL has taken steps to monitor maintenance activities at HFIR more closely [see, e.g., MMES, 1988j].

The committee believes that reactor safety would be better served if the reactor operations at ORNL had a small, in-house staff of maintenance personnel dedicated to reactor maintenance. This staff could then be trained and qualified specifically for work on the HFIR in order to ensure familiarity with the reactor and awareness of correct maintenance procedures and the consequences of maintenance errors. The director of the reactor and the maintenance manager would be responsible for evaluation of performance of individuals in this maintenance group. When the maintenance staff are not occupied with work on the reactor, they could still work on other tasks at ORNL, but their management and training should be optimized for reactor maintenance.

Larger jobs might still require individuals from a central laboratory work force or from outside the laboratory, but the work could then be conducted under direct surveillance and control by qualified in-house individuals.

Recommendation: The contractor should dedicate a small in-house staff with a range of expertise to reactor maintenance. This staff should be under the direct supervision of HFIR management. The HFIR training program should be expanded to include training for the newly dedicated staff of maintenance workers.[2]

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[2] While the report was going to press, the committee was informed that the contractor had recently made changes along the lines recommended in the report. The committee was not in a position to confirm these changes. The committee recommends that DOE and the contractor carefully monitor the new maintenance organisation in order to ensure that, indeed, a properly supervised, high-quality maintenance program has been established at HFIR and is now operating effectively.

Conclusion

The test and research reactors represent a significant national resource for scientific research, for the production of radioisotopes for medical and other purposes, and for advancing the knowledge of reactor technology. However, their capacity to continue to serve as tools for science, industry, medicine, and national defense will be jeopardized if the reactors are not operated in a way that comports with public expectations for reactor safety. Because several of the facilities were designed 20 to 30 years ago, they are confronting some expected, and some unexpected, problems of aging. Special and continuous activities to ensure safety are necessary. It is hoped that this report will offer helpful guidance to DOE in these efforts.

The committee also hopes that this report and its predecessor on the defense production reactors will be viewed in the proper context. The real challenge confronting DOE is not only to respond to specific issues that this committee or others have identified, but rather, it is to establish a viable, vigilant safety enterprise that is capable of identifying and correcting safety issues without extensive outside prodding. Perhaps this is the most significant message of these reports.

There are signs that the department and its contractors recognize this need. The DOE safety apparatus is improving and is striving to meet its obligations. In this connection, it is important to recognize that DOE itself requested the committee's scrutiny of the

Class A reactors and has embarked on extensive efforts to respond to the recommendations in the previous report. These steps are healthy signs. Nonetheless, the dedication to improving reactor safety cannot be episodic. There must be a vital, continuous, and long-term commitment to safety that permeates all levels of DOE and the contractor organizations. The relationship between the department and its contractors must be a partnership of common goals, complementary capabilities, and, most importantly, mutual dedication to the safety of the reactors and the protection of operating personnel and the public.

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Appendix A
Statement of Task

The National Research Council will undertake an assessment of safety and technical issues raised by the nuclear reactor accident at Chernobyl. The assessment will focus on the eleven Class A (over 20 MWT) reactors operated by the Department of Energy. These eleven reactors are the N Reactor at Hanford; the C, K, L, and P production reactors at Savannah River; the Fast Flux Test Facility at Hanford; the Experimental Breeder II and the Advanced Test Reactor at Idaho Falls; the High Flux Beam Reactor at Brookhaven; and the High Flux Isotope Reactor and Oak Ridge Research Reactor at Oak Ridge.

The committee of individuals expert in nuclear reactor safety, risk analysis and assessment, and management of large production and research programs, will carry out the review. The committee will obtain the results of DOE's ongoing safety assessments of production reactors and will receive briefings on what is known about the Chernobyl accident. In addition, the committee will respond to Secretary Herrington's request for public participation by holding public meetings to receive ideas and information from interested individuals and groups.

This Statement of Task is taken largely from the August 1986 contract between the U.S. Department of Energy and the National Academy of Sciences-National Research Council.

The committee will report within nine months on matters of immediate safety concern for DOE reactor design, construction, and operation. It is expected that the committee will consider a number of more generic, perhaps longer-term, safety issues that arise from a more complete understanding of the Chernobyl accident; these will be discussed in a later report.

Appendix B
Biographical Sketches of Committee Members

RICHARD A. MESERVE is a partner in the Washington law firm of Covington & Burling. He holds both a law degree from Harvard Law School and a Ph.D. degree in applied physics from Stanford University, where he did postdoctoral work on the theoretical properties of paramagnets and techniques to calculate molecular properties. In 1976, he was a clerk for Supreme Court Justice Harry A. Blackmun, and in 1977, he was appointed Legal Counsel and Senior Policy Analyst in the White House Office of Science and Technology Policy (OSTP). At OSTP he helped develop policies designed to promote the technological advance of American industry and conducted reviews of energy technology issues. In addition, he served as executive director of an interagency committee concerned with nuclear power plant safety. Mr. Meserve has been a member of several study committees of the National Research Council, including most recently the Panel to Study the Impact of National Security Controls on International Technology Transfer.

DAVID C. ALDRICH is a vice president at Science Applications International Corporation (SAIC). He has worked primarily on nuclear facility safety and waste management problems, and is an expert in radiological accident health and environmental and economic consequence evaluation. Prior to joining SAIC, Dr. Aldrich was supervisor of the Safety and Environmental Studies Division of Sandia National Laboratories, where he worked on a wide variety of reactor safety issues, including management of an NRC-sponsored program to develop a new set of risk assessment computer codes covering thermal-hydraulic behavior, fission product source terms, and offsite consequences of severe reactor accidents. He is a member of an International Atomic Energy Agency (IAEA) advisory group on emergency response decision making, and is active in the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA), having served as chairman of NEA's Group of Experts on Radiological Accident Consequences.

GEORGE APOSTOLAKIS is a professor in the Mechanical, Aerospace, and Nuclear Engineering Department of the University of California, Los Angeles. His research activities are in the development of methods for the assessment of risks from complex technological systems, nuclear reactor safety, and toxic waste disposal. He has published extensively on data analysis, human error modeling, and fire risk assessment. He has been a consultant on probabilistic risk assessment to private industry, government, national laboratories, and international organizations. He is a founding member and currently the president of the Southern California Chapter of the Society for Risk Analysis. He is also a vice president of the American Association for Structural Mechanics in Reactor Technology. He is coeditor of the international journal *Reliability Engineering and System Safety*.

RICHARD S. DENNING is senior research leader for nuclear safety at Columbus Laboratories, Battelle Memorial Institute. He is an expert in radioactive source term and severe accident research. His work focuses on reactor safety and risk, including core meltdown behavior, radionuclide transport, transient thermal hydraulics, and criticality and shielding analysis. He holds a Ph.D. degree in nuclear engineering from the University of Florida. He is currently a member of the DOE Advisory Committee on Nuclear Facility Safety.

RONALD GAUSDEN is currently a consultant on nuclear energy. He is the former chief inspector of the Nuclear Installations Inspectorate (NII) of the U.K. Health and Safety Executive. From 1960 to 1978, Mr. Gausden served in a number of supervisory positions at the NII. Prior to that, he was group manager at Windscale, where in 1957 a production reactor overheated during a Wegner energy release causing a graphite fire and release of radioactive particulates. He has authored papers for professional conferences on nuclear safety standards and nuclear power plant regulatory procedures, and in 1982 was awarded the C.B. for meritorious public service by Her Majesty the Queen.

DAVID L. HETRICK is professor of nuclear and energy engineering at the University of Arizona in Tucson. His research interests center on reactor dynamics and simulation. He is an administrative judge for the U.S. Nuclear Regulatory Commission, and has served as an IAEA technical expert on assignment to Mexico's Instituto de Investigaciones Electricas. Dr. Hetrick has also been a visiting professor of nuclear engineering at the University of Bologna, Italy. He has served as a consultant on reactor dynamics to government and industry and is the author of numerous articles on reactor physics and nuclear safety.

WILLIAM KASTENBERG is chairman of the Mechanical, Aerospace, and Nuclear Engineering Department at the University of California, Los Angeles. His research focuses on nuclear reactor safety, the development of risk-benefit and cost-benefit analysis, and environmental modeling for nuclear power installations. He has served as a senior fellow of the Advisory Committee on Reactor Safeguards, where he developed methods for applying probabilistic acceptance criteria to nuclear and nonnuclear technologies. He has been a consultant to a number of other governmental panels, including the President's Nuclear Safety Oversight Committee. He is the author of over 100 journal and proceedings publications relating to reactor safety and risk assessment, and recently served as a member of the National Research Council's Committee on Nuclear Safety Research.

HERBERT KOUTS is chairman of the Department of Nuclear Energy at Brookhaven National Laboratory. He was the first director of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research, having previously headed the Division of Reactor Safety Research at the Atomic Energy Commission. He is a former member and chairman of the Advisory Committee on Reactor Safeguards, and has been affiliated, either as a consultant or as a member, with a number of national and international groups focusing on reactor safety and safeguards, including the NRC's Risk Assessment Review Group, the President's Nuclear Safety Oversight Committee, the European-American Committee on Reactor Physics, the DOE Defense Energy Task Force, the American Nuclear Society (ANS) Special Committee on Source Terms, and several New York City and State advisory commissions on nuclear issues. He has also served on several advisory panels to review the safety of the N Reactor and the Fast Flux Test Facility. He is currently a member of the IAEA's International Nuclear Safety Advisory Group and the National Academy of Engineering, and is the recipient of several distinguished awards.

DAVID D. LANNING is professor of nuclear engineering at the Massachusetts Institute of Technology. He has been a consultant to a number of firms active in the electric utility and nuclear industries, including Stone and Webster Engineering Corporation, Northern States Power, Boston Edison, and GA Technologies. His professional interests include nuclear engineering education and the design, safety, control, and operation of nuclear reactor systems. In the 1950s he worked for General Electric at Hanford and in the 1960s for Battelle Northwest Laboratories. At MIT during the 1970s he was codirector of the MIT Research Reactor with the responsibility for the design and installation of the MIT R-II. He has also been a co-principal investigator of MIT's Program on Nuclear Power Plant Innovation in the area of modular high-temperature, gas-cooled reactors, and he is also the group coordinator for the Advanced Instrumentation and Control Program in the MIT Nuclear Engineering Department.

KAI N. LEE is associate professor at the Institute of Environmental Studies and in the Department of Political Science at the University of Washington. He is also a member of the Northwest Power Planning Council, having been appointed to the council by the governor of the state of Washington. He is a former member of the Office of Technology Assessment's advisory panel on radioactive waste disposal, and has served on a number of National Research Council committees, including past membership on the Environmental Studies Board and current membership on the Board on Radioactive Waste Management. His research interests include energy and environmental policy, regional power development, nuclear waste management, environmental conflict and dispute settlement, and the influence of technological change on American political life.

SALOMON LEVY is and has been the president of S. Levy Incorporated, a consulting firm to the power industry since 1977. He currently consults for many electric utilities and several power equipment manufacturers, the Electric Power Research Institute, two national laboratories, and the Research Division of the U.S. Nuclear Regulatory Commission. From 1953 to 1977, he was employed by the General Electric Company in various technical and managerial positions. He was successively manager, heat transfer and fluid flow development; manager, systems engineering; manager, design engineering; general manager, Nuclear Fuel Department; and general manager, Boiling Water Reactor System Department. In April 1975, he became general manager for Boiling Water Reactor Operations where he was responsible for all the engineering and manufacturing of General Electric's nuclear power business. Dr. Levy is a member of the National Academy of Engineering and a fellow of the American Society of Mechanical Engineers from which, in 1966, he received the ASME Heat Transfer Memorial Award. He is also a member of the American Nuclear Society, and in 1987 he received the ANS Thermal Hydraulics Division Technical Achievement Award. He is a director of Iowa Electric, a member of the oversight committees for four nuclear power plants, and a member of the U.S. Nuclear Regulatory Commission's Nuclear Safety Research Review Committee. He has authored more than 50 published technical papers.

DANA A. POWERS is supervisor of the Reactor Accident Source Terms Division of Sandia National Laboratories. Dr. Powers' particular research interests are the thermodynamics and kinetics of material processes under severe reactor accident conditions. He has worked extensively on core debris interactions with concrete and the behavior of radionuclides under accident conditions. He has served as a consultant to the Advisory Committee on Reactor Safeguards, the International Atomic Energy Agency's review of the Chernobyl accident, and the Rogovin Commission review of the Three Mile Island accident. He is currently a member of the DOE Advisory Committee on Nuclear Facility Safety.

HENRY E. STONE recently became a consultant after a 38-year career with General Electric (GE) in nuclear related activities. He was at Knolls Atomic Power Laboratory from 1950 to 1973 in various positions of reactor and plant design, construction, operation, and training and was general manager for the last six years. In 1974 he became manager of Operational Planning in the GE commercial nuclear business and in 1975 he became general manager of GE's Boiling Water Reactor Systems Department. In 1977 he became general manager of the Nuclear Energy Engineering Division, with responsibility for boiling water reactor engineering, engineered equipment procurement, and operation of the Vallecitos Nuclear Center. In the early 1980s he served on an NAS committee studying nuclear technology for space application and on a DOE safety panel on light-water reactors. He was elected as vice president of GE in 1978 and chief engineer in 1984. Mr. Stone is a member of the American Nuclear Society, a fellow in the American Society of Mechanical Engineers, and a member of the National Academy of Engineering.

THEO THEOFANOUS is professor of chemical and nuclear engineering and director of the Center for Risk Studies and Safety at the University of California in Santa Barbara. He has served as a consultant to the Advisory Committee on Reactor Safeguards since 1971 and has participated in a number of special Nuclear Regulatory Commission advisory/review committees and panels, most recently on one concerned with the peer review of NUREG-1150. He participated in the Nuclear Regulatory Commission's Task Force on the Chernobyl Accident and was a member of the U.S. delegation to the IAEA special meeting on Chernobyl. He is an editor of the *Journal of Nuclear Engineering and Design*. His research centers on thermal hydraulics and transport phenomena in turbulent and multiphase systems, with particular emphasis on nuclear and chemical reactor safety applications.

NEIL TODREAS is chairman of the Department of Nuclear Engineering at MIT, where he has been teaching since 1970. He serves as codirector of the MIT Summer Reactor Safety Program. Prior to joining the MIT faculty, he was senior reactor engineer for the Division of Reactor Development and Technology at the Atomic Energy Commission, where he served as lead engineer on the design of the core and reactor assembly of the Fast Flux Test Facility. Dr. Todreas is a member of the National Academy of Engineering and a fellow of the American Nuclear Society and the American Society of Mechanical Engineers, and he serves as the chairman of the Nuclear Safety Research Review Committee of the U.S. Nuclear Regulatory Commission. He is also on the editorial board of the *Journal of Nuclear Engineering and Design* and is the author of numerous technical articles and papers on reactor design methods and thermal hydraulics.

WILLIAM WEGNER is president of Basic Energy Technology Associates, Inc. (BETA), a small technical consulting group specializing in providing assistance to nuclear utilities in the management, design, and operation of nuclear power plants. For 15 years prior to forming BETA, Mr. Wegner was deputy director under Admiral Hyman G. Rickover of the U.S. naval reactors program. In 1980-1981 he served as a member of the staff of the "Crawford Committee," which conducted a review of the safety of DOE reactors.